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Validation of the Emergency Planning Basis for the Bruce Power Site

K-600240/RP/0019 R00

June 30, 2021

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Revision Summary

Rev	Date	Author	Comments
D00	June 2021	J. Morris, J. Kennedy, P. Maka	Draft issued for Bruce Power review.
R00	June 2021	J. Morris, J. Kennedy, P. Maka	Bruce Power review comments addressed.

Certification Statement

I, the undersigned, being a licensed professional engineer in the province of Ontario and being competent in the applicable field, have prepared or directly supervised the preparation of this document, following the procedures of the Kinectrics quality management system.

Kinectrics document and revision no. K-600240/RP/0019 R00

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EXECUTIVE SUMMARY

The Province of Ontario's Office of the Fire Marshal and Emergency Management (OFMEM) last revised the Provincial Nuclear Emergency Response Plan (PNERP) in 2017 to incorporate lessons learned following the Fukushima nuclear accident in Japan. The 2017 PNERP update was a significant undertaking as it was the first revision of the PNERP following the Fukushima accident. Key changes introduced as part of the 2017 PNERP included:

- Improved guidance on considerations for the potential consequences of Beyond Design Basis Accidents (BDBAs) in emergency planning; and
- Revisions to the size and terminology associated with emergency planning zones, including the introduction of a new Contingency Planning Zone.

Consistent with requirements for periodic review, OFMEM intends to issue a revised PNERP in 2022 to reflect industry developments and lessons learned since the 2017 PNERP update. This report is prepared to support Bruce Power with providing inputs to OFMEM as part of the upcoming revision of the PNERP. Specifically, the intent of this report is to document the analysis of off-site consequences for a range of accident scenarios which can be used to assist in validating the appropriateness of the current emergency planning zones surrounding the Bruce Power site. It is prudent to undertake this study given that:

- Previous analysis used as inputs to the PNERP planning basis is primarily based on off-site consequences resulting from generic or Darlington-specific accident scenarios which do not fully reflect post-Fukushima enhancements implemented at Ontario CANDU stations;
- Bruce Power has made progress on various post-Fukushima enhancements since the 2017 PNERP update such as the installation of Shield Tank Overpressure Protection on all units and the issuance of revised Severe Accident Management Guidance documentation; and
- Following the 2017 PNERP update, further guidance has been provided in relevant industry standards such as Canadian Standards Association (CSA) N1600, "General Requirements for Nuclear Emergency Management Programs" and CSA N290.16, "Requirements for Beyond Design Basis Accidents" regarding BDBA analysis, which is a key input to the PNERP planning basis.

This study utilizes a 'best estimate' approach in analyzing the station response to BDBAs, which is consistent with current industry best practices. The appropriateness of this approach is further supported by:

- The implementation of design features at Bruce Power stations which are specifically intended for use in BDBA conditions;
- The existence of procedures to perform necessary actions in support of the BDBA response; and
- Training activities which demonstrate the capability to effectively execute credited actions.

In order to confirm the validity of the existing emergency planning zone sizes surrounding the Bruce Power site, three accident scenarios were analyzed. These scenarios were selected to reflect representative scenarios which encompass the range of accidents considered in the PNERP planning basis. The scenarios analyzed in this study are summarized below.

- **Case 1:** a large Loss of Coolant Accident (LOCA) on a single unit where various safety systems function per design. This scenario was selected as a representative Design Basis Accident.
- **Case 2:** a LOCA on a single unit which progresses to a Severe Accident (SA) as a result of the failure of the Emergency Coolant Injection System and a loss of moderator cooling. This scenario was selected as a representative single unit BDBA which progresses to a SA.
- **Case 3:** a station blackout which results in a loss of Alternating Current (AC) power to all four units that progresses to a SA. This scenario was selected as a representative multi-unit BDBA which progresses to a SA.

In all cases, the estimated off-site doses are sufficiently low such that it is not anticipated off-site protective actions would be required. This illustrates the effectiveness of various post-Fukushima enhancements Bruce Power has implemented to mitigate the consequences of BDBAs. Furthermore, the results of the analysis indicate that the existing emergency planning zones are more than adequate to manage the off-site response to representative accident scenarios which form the PNERP planning basis. The existing emergency planning zones ensure there is margin available to manage the off-site response to very low probability, higher consequence SAs which have the potential to result in the need for off-site protective actions. Thus, the changes to the emergency planning zones surrounding the Bruce Power site introduced through the 2017 PNERP update have been confirmed to be, and continue to be, more than appropriate.

Acronyms and Abbreviations

AAZ	Automatic Action Zone	GC	Generic Criteria
ADDAM	Atmospheric Dispersion and Dose Analysis Method	HTS	Heat Transport System
AIM	Abnormal Incident Manual	IVR	In-Vessel Retention
BDBA	Beyond Design Basis Accident	LOCA	Loss of Coolant Accident
CANDU	CANada Deuterium Uranium	LRF	Large Release Frequency
CSA	Canadian Standards Association	MAAP	Modular Accident Analysis Program
CSP	Critical Safety Parameter	NERP	Nuclear Emergency Response Plan
DBA	Design Basis Accident	OFMEM	Ontario's Office of the Fire Marshal and Emergency Management
DPZ	Detailed Planning Zone	OIL	Operational Intervention Level
ECI	Emergency Coolant Injection	PARs	Passive Autocatalytic Recombiners
EMC	Emergency Management Centre	PNERP	Provincial Nuclear Emergency Response Plan
EME	Emergency Mitigating Equipment	SA	Severe Accident
EMEG	Emergency Mitigating Equipment Guideline	SACRG	Severe Accident Control Room Guide
EOC	Emergency Operations Centre	SAMG	Severe Accident Management Guidance
ERO	Emergency Response Organization	SP	Support Parameter

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1.0 INTRODUCTION

The Province of Ontario's Office of the Fire Marshal and Emergency Management (OFMEM) last revised the Provincial Nuclear Emergency Response Plan (PNERP) in 2017 to incorporate lessons learned following the Fukushima nuclear accident in Japan. In support of these updates, Bruce Power provided input to OFMEM relating to potential changes required to the planning basis in order to ensure an effective emergency response to Beyond Design Basis Accidents [1].

The 2017 PNERP update was a significant undertaking as it was the first revision of the PNERP following the Fukushima accident. Key changes introduced as part of the 2017 PNERP included:

- Improved guidance on considerations for the potential consequences of Beyond Design Basis Accidents in emergency planning;
- Revisions to the size and terminology associated with emergency planning zones, including the introduction of a new Contingency Planning Zone; and
- Clarification of the required frequency for periodic reviews of the PNERP.

Bruce Power intends to provide input to OFMEM as part of a planned update of the PNERP scheduled for 2022. The intent of providing this input is to provide data and analysis results for representative accident scenarios to assist in the process of validating the current emergency planning zones surrounding the Bruce Power site. It is prudent to perform this validation exercise as Bruce Power has made significant progress on post-Fukushima initiatives since the 2017 PNERP update.

Bruce Power has sought assistance from Kinectrics in performing a study to analyze the off-site consequences for a range of accident scenarios which are aligned with the PNERP planning basis.

2.0 OVERVIEW OF PROVINCIAL EMERGENCY PLANNING BASIS

The response to a nuclear emergency at the Bruce Power site requires a co-ordinated response from multiple organizations. This co-ordination is achieved through a series of integrated emergency plans. Key organizations who support the emergency response have dedicated emergency plans; plans which are of particular relevance to the emergency planning basis for the Bruce Power site are illustrated below in Figure 1.

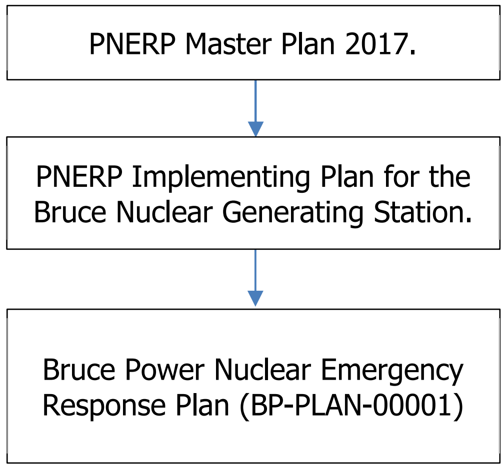

Figure 1: Hierarchy of Emergency Plans for the Bruce Power Site

Provincial Nuclear Emergency Response Plan (PNERP) Master Plan

The PNERP Master Plan is maintained by the Province of Ontario’s Office of the Fire Marshal and Emergency Management and establishes the overall principles, policies, basic concepts, organizational structures and responsibilities for Ontario’s response to a nuclear emergency.

Chapter 2 of the PNERP Master Plan defines the planning basis for nuclear emergencies. For the purposes of this study, the following excerpt from the PNERP Master Plan is of particular relevance [2]:

2.2.3 Reactor Facility Accidents

a) Nuclear emergency preparedness requires a planning basis which considers both design basis accidents and beyond design basis accidents (BDBAs) including multi-unit scenarios where applicable. For a detailed explanation regarding the basis for these reference accidents, refer to Annex L – PNERP Planning Basis Background.

b) While the planning basis should include a wide range of accidents, the amount of detailed planning should decrease as the probability of the accidents’ occurrence decreases. For this reason, the planning basis for managing a nuclear emergency must strike an appropriate balance.

c) Reactor facility safety analysis and risk assessments shall be used to inform the planning basis.

Annex L of the PNERP Master Plan contains a listing of various sources of information which were used as inputs to the PNERP planning basis. Specifically, Section 3.0 in Annex L of the PNERP Master Plan summarizes the results from an assessment which was performed to obtain insights into the potential off-site consequences resulting from BDBAs for CANDU reactors in Ontario [2]. Recognizing that there are design differences between the various CANDU reactors in Ontario (e.g., Bruce A/Bruce B, Pickering, and Darlington), a multi-unit BDBA scenario at the Darlington station was analyzed to provide general insights to aid in establishing the planning basis for each station.

The approach described above is appropriate given that the PNERP Master Plan is not intended to establish site-specific emergency planning bases. Additionally, the PNERP Master Plan identifies appropriate limitations on the use of this information in that it should not be used as the sole source of information for nuclear emergency preparedness activities. However, it is important to acknowledge that this assessment was based on modelling of the Darlington station prior to the Fukushima event. As a result, the contents of Section 3.0 in Annex L of the PNERP Master Plan do not reflect enhancement initiatives implemented by various utilities in response to the lessons learned from the Fukushima event. It is prudent to ensure post-Fukushima initiatives are reflected in the PNERP planning basis in order to reflect the significant enhancements made to nuclear emergency preparedness capabilities since the Fukushima event and the positive impact this has on the expected station post-accident response.

Provincial Nuclear Emergency Response Plan (PNERP) Implementing Plan

Each nuclear site which is subject to the PNERP Master Plan has a dedicated PNERP Implementing Plan. The purpose of the Implementing Plan is to provide clarity on the applicability of requirements in the PNERP Master Plan to a given nuclear site. Reference [3] documents the PNERP Implementing Plan for the Bruce Nuclear Generating Station[1] which addresses both the Bruce A and Bruce B stations.

Chapter 2 of the PNERP Implementing Plan describes the planning basis for nuclear emergencies at the Bruce Power site. It contains a description of a representative Design Basis Accident (DBA) scenario for the purposes of emergency planning along with general guidance on the characteristics of BDBAs.

The discussion of DBAs and BDBAs in the PNERP Implementing Plan reflects fundamental differences in the characteristics associated with these two types of accidents. Table 1 illustrates key differences between DBAs and BDBAs for the purposes of emergency planning.

[1] For simplicity, the PNERP Implementing Plan for the Bruce Nuclear Generating Station is referred to as the 'PNERP Implementing Plan' throughout the rest of the report.

Table 1: Comparison of Design Basis Accidents (DBAs) and Beyond Design Basis Accidents (BDBAs)

Accident Characteristic	DBA	BDBA
Frequency	• Event frequency between 1 in 100 reactor years and 1 in 100,000 reactor years [4].	• Event frequency is less than 1 in 100,000 reactor years [4].
Anticipated Consequences	• Releases of radioactive material are kept within authorized limits as the facility is designed to respond to DBAs.	• BDBAs may or may not result in fuel damage, depending on the success of various mitigating measures. The level of fuel damage correlates to the amount of radioactive material released. • Broad range in potential consequences associated with BDBAs. General relationship is that the severity of the consequences increases as the frequency of the event decreases.

Accident Characteristic	DBA	BDBA
Required Mitigating Actions	• Use of permanent plant systems; and • Use of event-specific response procedures.	• Initial efforts focus on attempting to make use of permanent plant systems. However, some permanent equipment is anticipated to be unavailable as a consequence of the event. • Use of Emergency Mitigating Equipment (EME) (e.g., pumps, generators) to prevent fuel damage. • Use of Severe Accident Management Guidance (SAMG) to mitigate fuel damage/protect containment. SAMG response will make use of EME as appropriate (e.g., actions may be taken to use EME pumps to supply cooling water to different connection points in comparison to the initial EME response).
Potential Variability in Outcomes	• Low variability in potential outcomes as there are procedures in place to respond to DBAs [5].	• Wider variability in potential outcomes due to uncertainty in event progression. • Multiple layers of defence reduce likelihood of lower frequency/higher consequence events [5].

The guidance contained in the PNERP Implementing Plan is appropriate for the purposes of characterizing the hazards which need to be addressed as part of the emergency planning basis for the Bruce Power site. However, the PNERP Implementing Plan does not explicitly discuss any additional analyses/assessments specific to the Bruce Power site which were used to define the planning basis. Thus, it is appropriate to perform additional site-specific analysis to aid in validating the contents of the PNERP Implementing Plan.

Discussion of Relevant Industry Standards

One of the primary objectives of the 2017 PNERP Master Plan was ensuring alignment with relevant industry standards and guidance documents. In the Canadian nuclear industry, the two industry standards which are of most relevance to emergency planning are:

- Canadian Nuclear Safety Commission REGDOC-2.10.1, "Nuclear Emergency Preparedness and Response"

  This document specifies requirements related to emergency preparedness programs that must be established in order for a facility to obtain/maintain an operating license. REGDOC-2.10.1 requirements are not discussed further in this report as they are not applicable for the purposes of establishing the off-site emergency planning basis. This is because the requirements in REGDOC-2.10.1 are not applicable to organizations (i.e., Province of Ontario's Office of the Fire Marshal and Emergency Management (OFMEM)) who are responsible for defining the off-site emergency planning basis.

- Canadian Standards Association (CSA) N1600, "General Requirements for Nuclear Emergency Management Programs"

  CSA N1600 outlines requirements for a comprehensive nuclear emergency management program. The target audience for this standard is any organization which may be involved in the emergency response at a nuclear facility. That is, the target audience for CSA N1600 is broader in comparison to the target audience for Canadian Nuclear Safety Commission REGDOC-2.10.1.

CSA N1600 is the Canadian industry standard which is of most relevance for future updates to the PNERP Master Plan and PNERP Implementing Plan. This is reflected in Section 1.0 in Annex L of the 2017 PNERP Master Plan [2] which indicates that the PNERP updates were informed by the release of a new edition of CSA N1600.

When the 2017 PNERP Master Plan was issued, the latest edition of CSA N1600 was the 2016 version of the standard. A revised version of CSA N1600 was issued earlier in 2021 and it is assumed that the 2021 edition of the standard will be used as an input to the 2022 PNERP Master Plan updates. Among other changes, the 2021 edition of CSA N1600 provides improved guidance on establishing the off-site emergency planning basis. Key points from the standard are summarized below [6]:

- The organization responsible for off-site planning (i.e., OFMEM) shall determine an appropriate planning basis in consultation with other organizations.
  - This requirement remains unchanged from previous versions of the standard and is consistent with the approach followed by OFMEM as part of the 2017 PNERP updates.
- Reactor facilities should consider the outputs of safety analyses for Beyond Design Basis Accidents (BDBAs) in accordance with CSA N290.16, "Requirements for Beyond Design Basis Accidents".
  - The current version of CSA N290.16 is the 2016 edition of the standard [7]. The 2016 edition is the first edition of the standard as CSA N290.16

  - was developed as part of the Canadian nuclear industry's response to the Fukushima accident and to establish a standardized industry approach for the treatment of BDBAs.
  - CSA N290.16 was developed in parallel with the 2016 update to CSA N1600. As a result, it was not possible to fully integrate CSA N290.16 content into CSA N1600 at that time. That is, the 2017 PNERP update does not make reference to CSA N290.16.

CSA N290.16 establishes requirements for the treatment of BDBAs in a variety of areas (e.g., determining design conditions, use of accident management guidelines). Key aspects of CSA N290.16 which are of particular relevance for the purposes of emergency planning are summarized below [7]:

- One of the primary objectives of BDBA management is to terminate an event before it progresses beyond In-Vessel Retention (IVR). IVR refers to states where core debris is retained in the reactor vessel and represents a significant transition point in terms of the potential consequences associated with a BDBA. If IVR cannot be maintained then the potential consequences of a BDBA increase significantly. Accordingly, industry efforts focus on maintaining IVR in order to mitigate the potential consequences of BDBAs.
- BDBAs should be analyzed using a 'best estimate' approach.

In the context of emergency planning, the goal of using a 'best estimate' approach is to establish a realistic plant response in order to ensure off-site consequences considered as part of the planning basis are not unnecessarily conservative. Examples of applying a 'best estimate' approach are described below:

- Systems which are anticipated to be unavailable as a result of the event should not be credited with operating (e.g., systems which are not qualified to operate under anticipated environmental conditions resulting from the accident, systems which require sources of electrical power or water that are expected to be unavailable).
- Mitigating measures which have been designed specifically for BDBA conditions should be credited with functioning as intended. Examples of measures which are appropriate to credit include:
  - Deployment of portable equipment such as pumps and electrical generators;
  - Operation of design features which do not require operator action (e.g., pressure relief from open vent lines which prevents failure of containment structures);
  - Operator actions which are documented in emergency response procedures/guidelines where areas of interest inside the plant are expected to remain accessible based on post-accident environmental conditions.

It is assumed that the updates to the 2022 PNERP Master Plan will ensure alignment with the 2021 edition of CSA N1600, including relevant guidance provided in CSA N290.16.

Summary of Emergency Planning Basis for the Bruce Power Site

Key points identified from the review of relevant emergency plans and industry standards are summarized below:

- There is a well-established hierarchy of emergency plans which enables Bruce Power, OFMEM, and other organizations to provide an integrated response to a nuclear emergency at the Bruce Power site.
- The existing emergency planning basis in the PNERP includes consideration of the consequences of Design Basis Accidents and BDBAs which is consistent with current industry best practices (e.g., 2021 edition of CSA N1600).
- The current sizing of the emergency planning zones reflects a conservative approach given that previous estimates of off-site consequences are primarily based on scenarios which do not reflect enhancements made by Bruce Power as part of its post-Fukushima response.
- Given the factors outlined above, it is prudent to analyze representative Bruce Power-specific scenarios for inputs to the PNERP planning basis. This exercise will aid in validating the appropriateness of the existing emergency planning zones for the Bruce Power site.

2.1 Emergency Planning Zones for the Bruce Power Site

The emergency planning zones surrounding the Bruce Power site reflect the potential off-site doses resulting from a range of Design Basis Accidents and BDBAs, along with local geographical features and population densities. Consistent with the Provincial Nuclear Emergency Response Plan (PNERP) Master Plan, there are four emergency planning zones in the area surrounding the Bruce Power site. Table 2 summarizes key information regarding the size and intent of the various emergency planning zones. Figure 2 and Figure 3 provide visual illustrations of the various emergency planning zones.

Table 2: Summary of Emergency Planning Zones

Emergency Planning Zone	Zone Size per PNERP Implementing Plan [3]	Intent of Planning Zone
Automatic Action Zone (AAZ)	3 km	• If required, protective actions would be initiated prior to a release. • Definition of the AAZ reflects that individuals in this zone would receive the highest doses following an accident due to their proximity to the Bruce Power site. • Use of automatic protective actions represents a conservative approach with minimal negative impacts given that AAZ population is lower in comparison to other emergency planning zones.
Detailed Planning Zone (DPZ)	10 km	• If required, protective actions would be initiated prior to a release. • Designation of DPZ reflects the need for detailed planning given the potential magnitude of off-site consequences for Design Basis Accidents and Beyond Design Basis Accidents. • Distinction between DPZ and AAZ reflects reduction in potential doses to the public with increasing distance away from site.

Emergency Planning Zone	Zone Size per PNERP Implementing Plan [3]	Intent of Planning Zone
Contingency Planning Zone	10 km – 20 km	• If required, protective actions would be initiated prior to a release. • Designation of a Contingency Planning Zone provides flexibility in responding to off-site consequences for Beyond Design Basis Accidents, recognizing potential for variability in the outcome of these events. • Use of contingency plans is consistent with the principle in the PNERP Master Plan that the amount of detailed planning should decrease as the probability of the accident decreases.
Ingestion Planning Zone	50 km	• If warranted, protective actions would be initiated following a release. • Designation of the Ingestion Planning Zone defines the range of post-accident monitoring activities required to protect the public from doses resulting from potential contamination of drinking water, or plant and animal products.

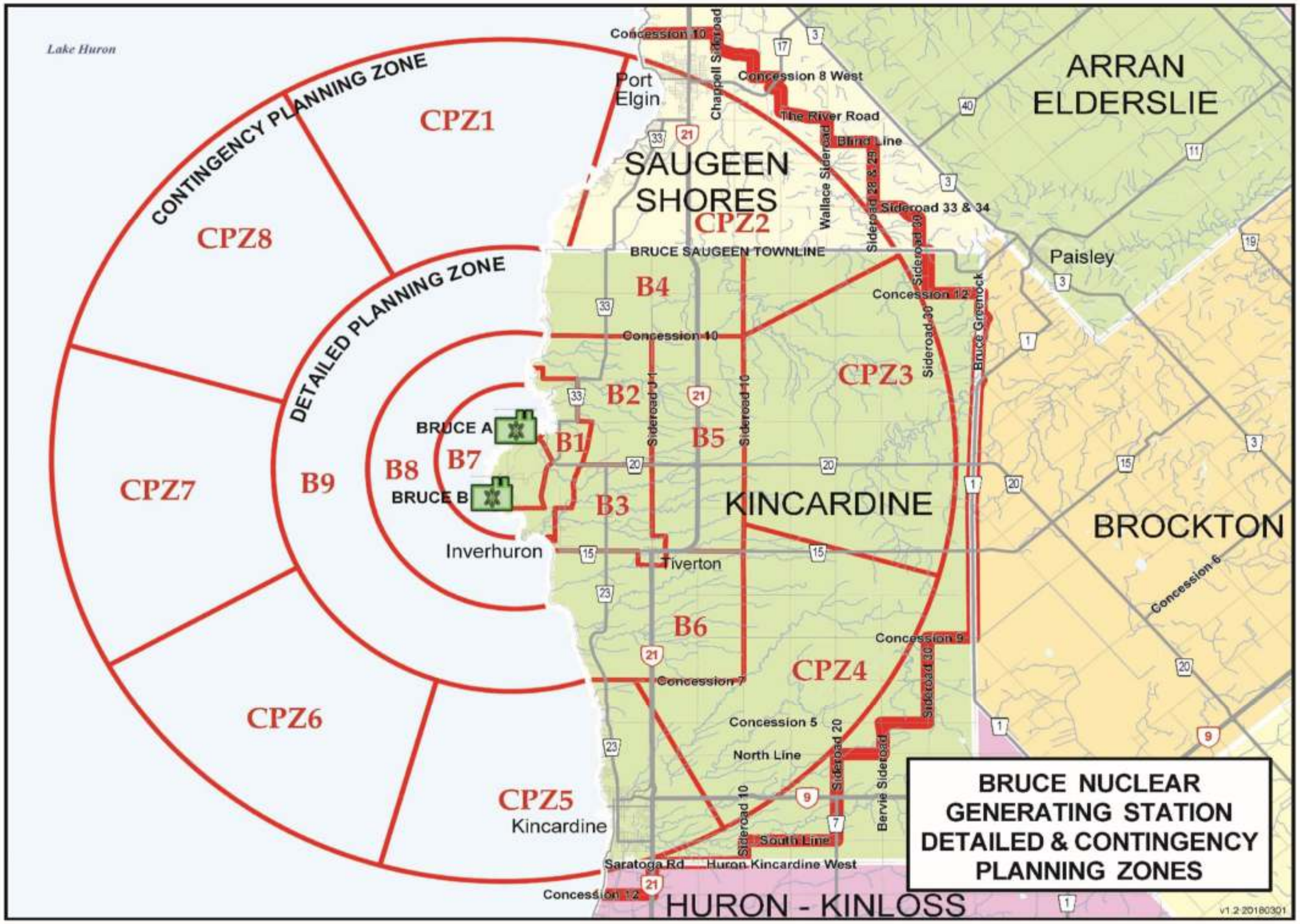

Figure 2: Detailed and Contingency Planning Zones [3]

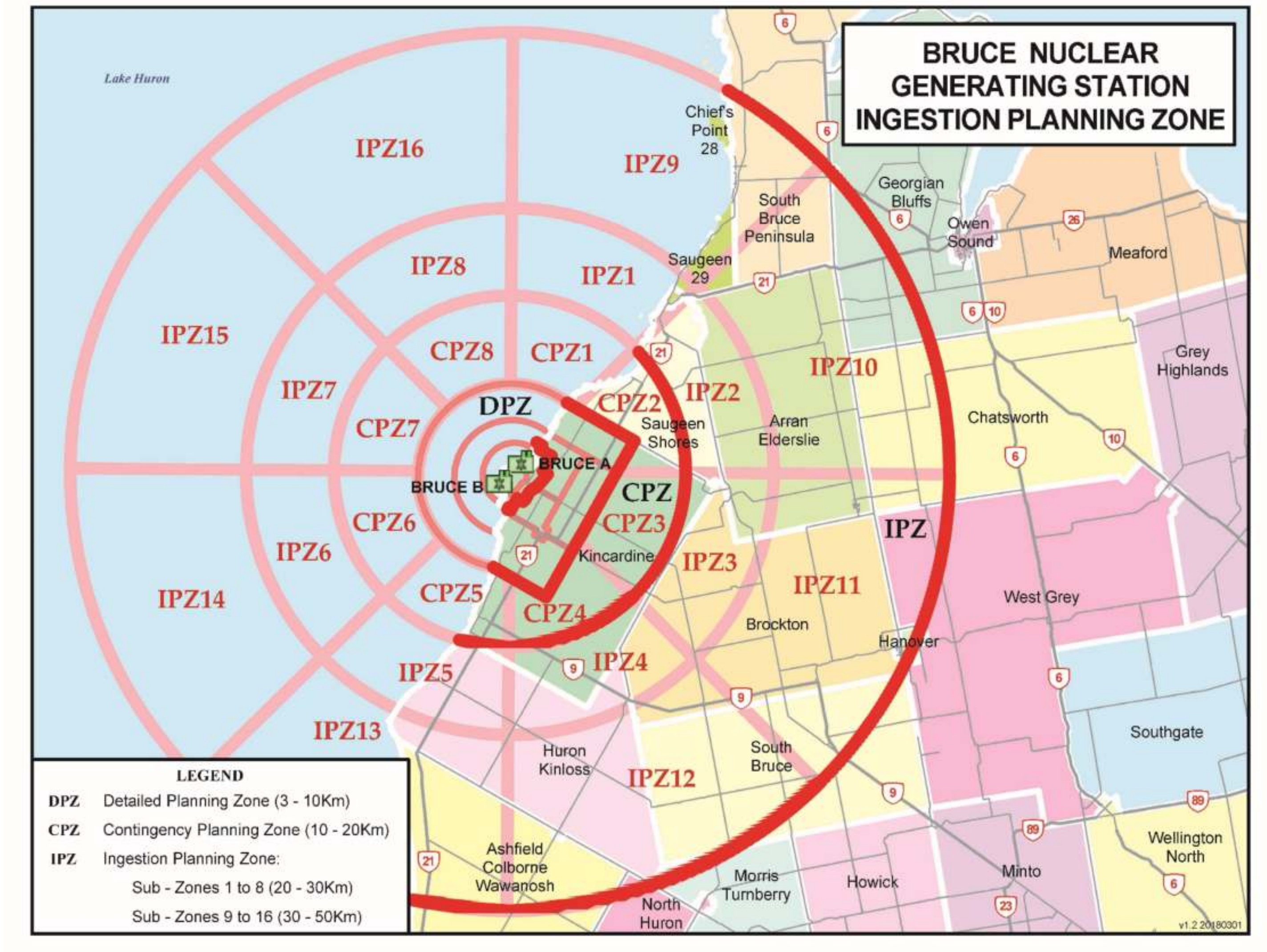

Figure 3: Ingestion Planning Zone [3]

2.1.1 Approach for Assessing the Need for Off-Site Protective Actions

The decision-making process for determining the need for off-site protective actions is based on comparing radiation doses (actual or modelled) to internationally accepted Generic Criteria (GC) or Operational Intervention Levels (OILs). In the early stages of an accident, dose projections based on current plant conditions are the primary source of information used for determining the need for off-site protective actions. This represents the best available information in advance of a release to the environment. Outputs from dose projections are compared to GC, which are reference levels that have been identified to protect the public from potential exposure to radiation. If the projected dose exceeds a GC, this indicates that the applicable protective action should be recommended for implementation in the affected emergency planning zone(s). Table 3 summarizes relevant information in the Provincial Nuclear Emergency Response Plan Master Plan regarding GC and the corresponding protective action strategies [2].

Table 3: Summary of Generic Criteria

Projected Dose	Protective Action Strategy	Additional Information
50 mSv (5 rem) thyroid dose in the first 7 days	Iodine Thyroid Blocking	• Intent of this strategy is to reduce/prevent the absorption of radioiodine by the thyroid gland. • This is accomplished by having individuals in affected zone(s) consume potassium iodide pills in advance of a release.
10 mSv (1 rem) whole body dose in the first 2 days	Sheltering	• Strategy utilizes shielding properties of buildings and their potential for ventilation control to reduce the radiation dose to people inside the buildings. • Typically used as a protective action if there is insufficient time to safely evacuate an area or if the risks of evacuation are higher than shelter-in-place (e.g., severe weather inhibits safe evacuation).

Projected Dose	Protective Action Strategy	Additional Information
100 mSv (10 rem) whole body dose in the first 7 days	Evacuation	• Dose rates in the affected zone(s) are expected to be sufficiently high for an extended period of time such that evacuation of the area is warranted to reduce/avoid exposing the population to the projected dose.
1 mSv (100 mrem) per year for ingestion of any one of the applicable food/beverage categories. **Note:** In this study, Derived Response Levels were used to determine if this GC would be exceeded. Additional information is provided in Table 4.	Restriction of distribution and ingestion of potentially contaminated: • Drinking water; • Milk; • Other foodstuffs and beverages.	• Avoid longer-term dose to individuals resulting from ingestion of food/beverages which have been contaminated as a result of a previous release to the environment.

To determine whether the fourth criterion above is met, Derived Response Levels are calculated which represent concentrations (in Bq/m^2) of radionuclides on the ground that could result in the consumption of food products over the course of an entire year exceeding the dose criterion (1 mSv per year). These Derived Response Levels are summarized below in Table 4.

Table 4: Derived Response Levels [8]

Foodstuff/Beverage	Radionuclide	Derived Response Level (Bq/m^2)
Root Vegetables	Cs-137	1.94E5
Leafy Greens	Cs-137	4.10E4
Grains	Cs-137	1.67E4
Milk	I-131	1.23E6

Following a release, the decision-making process for off-site protective actions relies less on dose projections and more on actual, measured levels of radioactivity in the field. Measured values obtained from various monitoring activities are compared to OILs in order to determine the need for off-site protective actions. If an OIL is exceeded, this indicates that the corresponding protective action should be implemented in the affected planning zone(s). The intent of the OILs is the same as the GC in that they represent reference levels which have been identified to protect the public from potential exposure to radiation. The overall decision-making process for off-site protective actions remains unchanged pre-release vs. post-release, as the OILs are derived from the GC in order to ensure consistency between the two approaches.

2.2 Responsibilities for On-Site and Off-Site Response

The response to an emergency requires a combination of on-site and off-site actions performed by various organizations. A collection of emergency plans, as discussed in Section 2.0, ensures an integrated approach between various organizations involved in the emergency response.

Bruce Power routinely conducts drills and exercises with many off-site organizations to ensure effective interoperability among all agencies during an emergency. However, Bruce Power's interoperability with the Province of Ontario's Office of the Fire Marshal and Emergency Management, the Provincial Authority for the off-site response, is of the utmost importance for the purposes of this study. Responsibilities for each organization are clearly defined in the Provincial Nuclear Emergency Response Plan (PNERP) Master Plan [2] and PNERP Implementing Plan [3] and can be simplified as follows:

- Bruce Power is responsible for on-site actions required to terminate the accident progression, which includes sending necessary notifications to organizations responsible for the off-site response;
- The Province of Ontario's Office of the Fire Marshal and Emergency Management is responsible for the overall off-site response, which includes identifying the need for off-site protective actions.

3.0 OVERVIEW OF BRUCE POWER'S EMERGENCY RESPONSE CAPABILITIES

Bruce Power's Nuclear Emergency Response Plan (NERP) is documented in Reference [9]. The purpose of the NERP is to describe the concepts, structures, roles and processes needed to implement and maintain Bruce Power's capability to prepare for and respond to a nuclear radiological emergency. This plan complies with requirements contained in Canadian Nuclear Safety Commission REGDOC-2.10.1 and ensures Bruce Power fulfills its responsibilities as defined in the Provincial Nuclear Emergency Response Plan (PNERP) Master Plan and PNERP Implementing Plan for the Bruce Nuclear Generating Station.

The NERP reflects an 'all hazards' approach to the planning and response to emergencies at the Bruce Power site. That is, the NERP is not based on a single type of event. Instead, it reflects capabilities which are required to be able to respond to a range of events that have been determined to be applicable to the Bruce Power site. More broadly, the NERP reflects requirements to support a sustained response to a Beyond Design Basis Accident, multi-unit incident involving a Severe Accident, resulting in an extended loss of off-site power for up to 72 hours without external assistance. Key capabilities outlined in the NERP are summarized below:

<u>Initial Event Response</u>

Each station is operated in accordance with procedures to address deviations from normal operating conditions and if necessary, respond to Design Basis Accidents. These procedures also include steps for Operations staff to characterize the nature of the event to determine the required level of emergency response. Characterizing the nature of the event ensures additional staff are mobilized in a timely manner to provide the necessary support in responding to an emergency. Characterizing the event involves the following steps:

- Classification
    - The classification of the event determines if the event is a station emergency such that the mobilization of an emergency response is required.
    - Declaration of a station emergency results in the activation of Bruce Power's Emergency Response Organization.
- Categorization
    - The categorization of the event determines the level of off-site response which is required. Specifically, the event categorization determines the required response from provincial and municipal organizations.
- Notification
    - The classification and categorization activities result in Operations staff sending notifications to mobilize staff required to support the emergency response. This includes Bruce Power staff who form part of the Emergency Response Organization as well as provincial and municipal organizations.

Emergency Response Organisation (ERO)

Bruce Power's ERO consists of two main groups: the shift ERO and the on-call ERO. Staff in both organizations are required to possess qualifications specific to their role and receive regular training to ensure they remain capable of effectively performing their role. Once activated, staff will perform their duties in accordance with role-specific emergency response procedures.

The shift ERO consists of on-site staff who support day-to-day station operations and are readily available to respond to an emergency. Staff who have key roles in the shift ERO will report to the Emergency Operations Centre (EOC) within 15 minutes of receiving a notification. Staff in these roles form part of the minimum shift complement, which ensures there are sufficient staff available 24/7 to fill all roles in the shift ERO. The shift ERO is structured to be able to manage the on-site and off-site response in the early stages of the event until the on-call ERO is declared operational.

Staff in the on-call ERO receive notifications at the discretion of the Shift Emergency Controller. At the request of the Shift Emergency Controller, the Emergency Shift Assistant will send out a notification to the on-call ERO duty if an event is of a serious enough nature to possibly require additional support. Staff in the on-call ERO are not required to be on-site at all times. However, there are staff on-call 24/7 who will report to the Emergency Management Centre (EMC) and be capable of becoming operational at the EMC within 90 minutes of receiving a notification.

In comparison to the shift ERO, the on-call ERO is significantly larger in size. This reflects the responsibility of the on-call ERO to assume responsibility for Centre of Site personnel, providing support to the incident station, communications to all external stakeholders and the co-ordination of the overall emergency response. These responsibilities include continuing to co-ordinate with the shift ERO to provide the necessary resources to support the on-site response while ensuring Bruce Power fulfills its responsibilities related to the off-site emergency response.

Facilities

There are dedicated facilities in place which staff who are part of the emergency response will report to. The primary facilities of interest for managing the on-site emergency response are the Bruce Power EOC and the EMC.

The Bruce Power EOC is a station facility where the initial, centralized coordination of the on-site and off-site response takes place. Staff who are part of the shift ERO will report to the EOC. There are separate EOCs at Bruce A and Bruce B which are both equipped with a range of communications equipment and are supported by back-up electrical power supplies. In the event that a station's EOC is unavailable, there is an alternate EOC location that is prepared to act as the back-up EOC at each station. If this is not a safe location to be used then the Mobile Operations Centre can be deployed to the station for use as the back up EOC and finally the other station's EOC can be used as an alternate work location. The Mobile Operations Centre can be used as an alternate command or support facility if required during any event at the request of the Shift Emergency Controller.

The EMC is used to co-ordinate and manage the overall Bruce Power site response to a nuclear emergency and provides the primary contact for communications with the provincial, federal, and local municipal EOCs and stakeholders. Staff who are part of the on-call ERO will report to the EMC, which is located off-site at the Bruce Power Visitors' Centre. Similar to the EOC, the EMC is equipped with a range of communications equipment and back-up electrical power supplies to sustain continued operations in the absence of off-site power. In the event that the EMC is unavailable, there are alternate locations in Kincardine or Port Elgin which staff can report to.

Mutual Aid and External Agreements

Bruce Power has pre-established arrangements in place with various organizations who directly support the emergency response as well as external organizations who may be able to provide technical support to Bruce Power in the event of an emergency. Key arrangements include, but are not limited to:

- Agreements exist with local fire departments for onsite firefighting support.
- Arrangements and procedures exist for local paramedic services and hospital support.
- Mutual aid agreements are in place with Canadian nuclear operators such as Ontario Power Generation. These agreements cover broad support which may be required during an emergency, ranging from technical expertise to the provision of materials/equipment.

Drills and Exercises

Bruce Power validates the NERP and implementing procedures through a series of emergency drills and exercises. The Bruce Power Drills and Exercises program includes performance objectives and criteria that test the implementation and functionality of plans, procedures, equipment, and the ERO as a whole. Many of the drills and exercises are focused on on-site response capabilities. However, the Drills and Exercises program also specifies requirements for Bruce Power to regularly conduct drills and exercises with the local municipalities and the Province of Ontario's Office of the Fire Marshal and Emergency Management to ensure effective interoperability among all off-site agencies and site emergency operations. Designated staff are assigned to evaluate the emergency response against pre-defined objectives and performance criteria specific to the drill/exercise scenario. Observations from evaluation activities are used to identify findings, opportunities for improvement, strengths, and good practices which are used to support continuous improvement of Bruce Power's emergency response capabilities.

Maintenance and Testing of Facilities and Equipment

The NERP relies on sufficient emergency facilities and equipment being available at all times. Bruce Power has a dedicated program, the Equipment Important to Emergency Response program, which is used to manage the availability of emergency response facilities and equipment. Emergency Management staff monitor, periodically test, and maintain emergency response facilities and equipment to ensure sufficient facilities/equipment are available at all times. For example, the availability of Emergency Mitigating Equipment is managed in accordance with this program and regular tests are performed to demonstrate Emergency Mitigating Equipment remains capable of operating as intended.

3.1 Post-Fukushima Enhancements

In response to lessons learned from the Fukushima accident, Bruce Power has implemented a wide range of enhancements to improve the capability of the Bruce A and Bruce B stations to respond to an extended loss of AC power due to a natural disaster. Relevant initiatives include a combination of emergency preparedness improvements and design modifications.

The discussion of Bruce Power's emergency response capabilities in Section 3.0 addresses current capabilities which reflect post-Fukushima enhancements. For example, constructing the Emergency Management Centre (EMC) off-site was a conscious decision made based on insights from the Fukushima accident. Lessons learned from Fukushima were also used as inputs for identifying and selecting equipment required to support EMC operations. A particular focus was placed on ensuring adequate provisions were made for communications equipment and back-up electrical power supplies to support EMC operations.

The remainder of this section provides an overview of significant design modifications implemented by Bruce Power, or which Bruce Power has committed to implementing, in order to enhance the capability of Bruce A and Bruce B to respond to Beyond Design Basis Accidents (BDBAs).

Emergency Mitigating Equipment (EME)

In the event that efforts to re-establish sources of on-site electrical power are unsuccessful, the deployment of EME will be requested. EME is stored at a dedicated off-site facility which is located across the road from the Bruce Power Visitors' Centre. The location of the storage facility was selected to ensure EME can be deployed on site in a timely manner while reducing the likelihood that the availability of EME is impacted by the events occurring on site.

EME consists of pumper trucks and portable generators which are deployed to supply electrical power to critical plant equipment and ultimately provide the capability to re-establish fuel cooling in each reactor and the Irradiated Fuel Bays. In addition to the pumper trucks and portable generators, the EME storage facility also contains payloaders which can be used to clear any debris from the roads leading to site or the incident station.

EME is operated by trained emergency personnel and deployed in accordance with approved procedures. There are pre-designated connection points at each station to connect the portable generators and supply water to the steam generators, Heat Transport System, moderator, or shield tank. EME is intended to prevent fuel damage such that a Severe Accident does not occur. However, it can also be used to support actions recommended through the application of Severe Accident Management Guidance.

Passive Autocatalytic Recombiners (PARs)

One of the lessons learned from Fukushima was the potential for hydrogen to accumulate inside containment, resulting in a potential flammability hazard. To mitigate the potential for hydrogen to accumulate inside containment following Design Basis Accidents (DBAs) or BDBAs, Bruce Power has installed PARs in all operating units. PARs do not require electrical power or operator intervention. They will automatically operate in the event that containment hydrogen concentrations exceed a minimum threshold which is significantly below levels where hydrogen flammability would be a concern. PARs support efforts to maintain the integrity of containment, which ultimately reduces potential doses to the public resulting from an accident.

Shield Tank Overpressure Protection

Severe Accident (SA) conditions will result in the generation of significant quantities of steam due to boiling off of water used to remove decay heat generated by each reactor. The generation of steam introduces the potential for containment pressures to reach values beyond what would be anticipated following DBAs such that design features which provide pressure relief capabilities for DBAs are potentially inadequately sized for BDBA conditions.

The capability to relieve pressure is of particular importance for the shield tank which surrounds the reactor vessel. In order to mitigate the potential for failure of the shield tank during a SA due to over-pressurization, Bruce Power has installed pressure relief provisions in all units. These provisions are referred to as Shield Tank Overpressure Protection, which consists of a pipe that discharges liquid/steam build-up from the shield tank into the larger containment structure. This feature allows for the full use of shield tank water to provide heat removal capabilities and ensures that low-pressure EME makeup water can be supplied to replenish shield tank inventory if required.

Containment Filtered Venting System (CFVS)

Existing filtered venting systems at Bruce A and Bruce B were designed to mitigate the consequences of DBAs. As a result, there are limitations on the manner in which these systems can be used in responding to BDBAs. Although the existing filtered venting system can be powered by portable generators, the primary area of concern is that multi-unit BDBAs have the potential to result in containment pressures beyond the capabilities of existing filtered venting systems. A capability assessment for the Emergency Filtered Air Discharge System has been performed to establish limitations on the use of this system as part of the BDBA response [10].

Bruce stations do not currently have a dedicated filtered venting system for BDBA conditions. However, Bruce Power is currently in the process of completing design work for a CFVS and has made commitments to the Canadian Nuclear Safety Commission to install a CFVS at both stations. The CFVS is designed specifically for use in responding to BDBAs and does not require electrical power to operate the system. Operator action is required to manually open valves to place CFVS in service. However, once in service, CFVS operates passively as it relies on the elevated containment pressure as the driving force for venting to the environment. As work on this initiative remains in progress, the current study does not credit CFVS. However, once installed, CFVS will result in further reductions to public doses resulting from certain multi-unit SAs.

4.0 OBJECTIVES

The objective of this study is to validate the appropriateness of the existing emergency planning zones surrounding the Bruce Power site, as identified in the 2017 Provincial Nuclear Emergency Response Plan (PNERP) Implementing Plan [3]. It is prudent to perform this study given that:

- Previous analysis used as inputs to the PNERP planning is primarily based on off-site consequences resulting from generic or Darlington-specific accident scenarios.
- Previous analysis does not reflect post-Fukushima enhancements that Bruce Power has implemented to mitigate the consequences of Beyond Design Basis Accidents. Additionally, at the time Bruce Power provided inputs in support of the 2017 PNERP update, some post-Fukushima enhancements were not fully implemented.

The objectives of this study are achieved by addressing the following four questions for each of the scenarios within the scope of the study:

1. Are there any areas which are projected to exceed 10 mSv effective dose in the first 2 days outside the currently established Detailed Planning Zone (DPZ)?
    - Addressing this question confirms whether the DPZ is adequately sized to address scenarios where the Province of Ontario's Office of the Fire Marshal and Emergency Management (OFMEM) would be expected to recommend members of the public to shelter-in-place.
2. Are there any areas which are projected to exceed 100 mSv effective dose in the first 7 days outside of the currently established Automatic Action Zone (AAZ)?
    - Addressing this question confirms whether the AAZ is adequately sized to address scenarios where OFMEM would be expected to recommend evacuation of certain areas in the immediate vicinity of the Bruce Power site.
3. Are there any areas which are projected to exceed 50 mSv adult thyroid dose in the first 7 days outside of the currently established DPZ?
    - Addressing this question confirms whether the DPZ is adequately sized to address scenarios where OFMEM would be expected to recommend members of the public to ingest potassium iodide pills.
4. Are any areas projected to exceed 1 mSv per year from the ingestion of root vegetables, leafy greens, grains, or milk outside of the currently established Ingestion Planning Zone?
    - Addressing this question confirms whether the Ingestion Planning Zone is adequately sized in terms of the geographical areas where surveys would be performed following a release to the environment to determine whether ingestion control measures are required post-release.

Figure 4 displays these the Generic Criteria doses alongside some common radiological dose examples to provide context on the magnitude of the various dose limits.

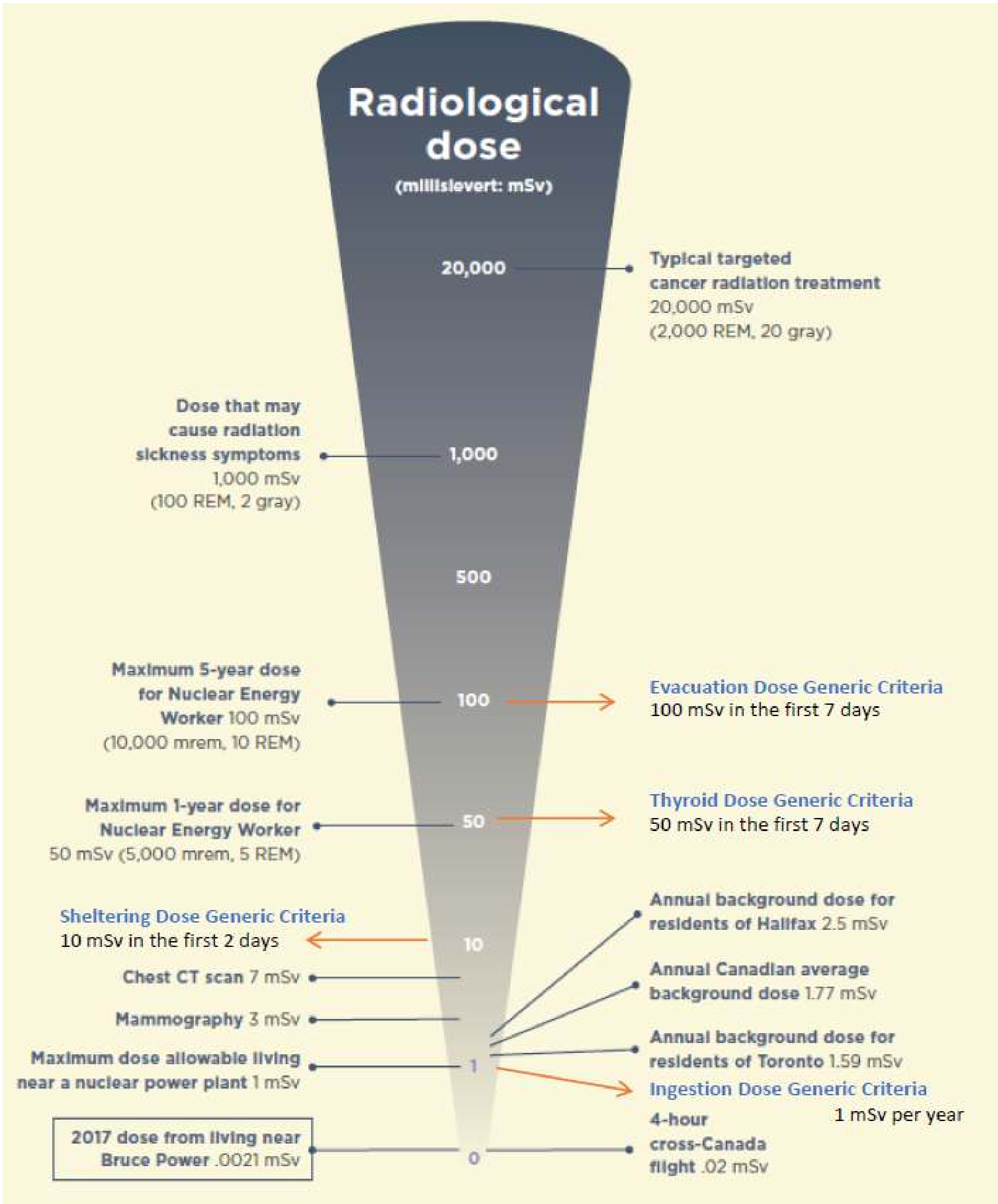

Figure 4: Radiation Dose Examples

5.0 METHODOLOGY

The validity of the emergency planning zones is demonstrated by analyzing a combination of Design Basis Accident and Beyond Design Basis Accident (BDBA) scenarios, consistent with the planning basis in the Provincial Nuclear Emergency Response Plan (PNERP) Master Plan. Particular focus is placed on the analysis of BDBA scenarios as by definition, BDBAs are expected to result in higher off-site doses than DBAs.

The analysis of BDBA scenarios utilizes a 'best estimate' approach, consistent with current industry best practices documented in Canadian Standards Association (CSA) N290.16, "Requirements for Beyond Design Basis Accidents" and the 2021 edition of CSA N1600, "General Requirements for Nuclear Emergency Management Programs". The current guidance in these standards was not available at the time of the 2017 PNERP update. Thus, the use of a 'best estimate' approach for the analysis of BDBA scenarios is appropriate given that it is assumed the standards referenced above will be utilized as inputs to the 2022 PNERP update.

Section 5.1 provides the basis for the accident scenarios which were analyzed as part of this study. Section 5.2 provides additional details on the tools used to perform the analysis.

5.1 Selection of Accident Scenarios

All scenarios analyzed in this study correspond to hypothetical events at the Bruce B station. Analysis of corresponding scenarios at the Bruce A station has not been performed as the PNERP Implementing Plan does not distinguish between accidents at the Bruce A and Bruce B stations. That is, the same emergency planning zones apply to both stations.

A total of three accident scenarios are analyzed, one DBA scenario (Case 1) and two BDBA scenarios (Cases 2 and 3). Key features of these scenarios and the basis for their inclusion in the study are provided in Table 5.

Table 5: Description of Accident Scenarios

Case #	Accident Description	Justification for Inclusion in Study
1	Loss of Coolant Accident (LOCA) • All post-accident mitigating systems function per design. • Single unit event.	• Representative scenario for Design Basis Accidents, as indicated in Provincial Nuclear Emergency Response Plan Implementing Plan [3].
2	LOCA/Loss of Moderator Cooling/Loss of Emergency Coolant Injection • Accident progression terminated through use of Emergency Mitigating Equipment (EME). • Single unit event.	• Representative scenario for a single unit Beyond Design Basis Accident (BDBA). • Scenario progresses to a Severe Accident (SA), which involves fuel failures. • Analysis of this scenario ensures consequences of single unit events are considered. This is important because single unit BDBAs may progress more rapidly in comparison to multi-unit BDBAs.
3	Station Blackout affecting all 4 Units • Accident progression terminated through use of EME. • EME make-up is established to maintain In-Vessel Retention.	• Representative scenario for a multi-unit BDBA. This also represents the most challenging scenario based on BDBA response objectives identified in CSA N290.16. • Scenario progresses to a SA, which involves fuel failures in all 4 units. • Analysis of this scenario ensures consequences of multi-unit events are considered. This is important because multi-unit events have the potential to result in more fuel failures and higher containment pressures in comparison to single unit events.

5.2 Description of Analysis Tools

The analysis of each scenario involves two steps:

1) Estimation of the source term; and
2) Analysis of off-site doses resulting from the release of the source term to the environment.

The use of analysis tools for the Design Basis Accident and Beyond Design Basis Accident scenarios is summarized below.

Design Basis Accident

No new analysis was performed to generate the source term. Previous work was performed in 2018 to generate the source term for a single unit large break Loss of Coolant Accident (LOCA) at Bruce B with all safety systems functional. This source term is documented in Reference [11] and was re-used for this study as it corresponds to the accident conditions described in Case 1.

The analysis of off-site doses was performed using the Atmospheric Dispersion and Dose Analysis Method (ADDAM) code. Section 5.2.2 contains additional details on the capabilities of the ADDAM code.

Beyond Design Basis Accident

The Modular Accident Analysis Program (MAAP) code was used to estimate the source terms for Cases 2 and 3. Section 5.2.1 contains additional details on the capabilities of the MAAP code.

The analysis of off-site doses was performed using the ADDAM code. Section 5.2.2 contains additional details on the capabilities of the ADDAM code.

5.2.1 MAAP-CANDU

MAAP-CANDU is an industry standard code for modelling the response of a CANDU reactor to a Severe Accident, including the performance of various systems such as the reactor core, steam generators, containment system, etc. It is an integrated system analysis code that is capable of modelling accidents initiated by LOCAs and a loss of heat sinks event initiated by a Station Blackout. It simulates, in addition to other physical and chemical processes, fuel temperature transients, damage to the core structures (e.g., fuel and reactor vessel), and fission product releases associated with a given accident.

MAAP5-CANDU was used to simulate the accident progression of Cases 2 and 3. The MAAP5-CANDU qualification report [12] documents all qualification activities which have been performed and concludes that MAAP5-CANDU is qualified for use for Severe Accident analysis. Thus, MAAP5-CANDU is qualified for use in applications such as this study.

The following outputs from MAAP5-CANDU were used as inputs to the ADDAM analysis:

- Release activity data; and

- Release mass and energy data.

Section 5.2.2 provides additional details on the methodology used to perform the ADDAM analysis in order to estimate the off-site doses associated with a given accident scenario.

5.2.2 ADDAM

5.2.2.1 Code Capabilities

ADDAM is the industry standard code used to estimate public doses following the release of radionuclides to the atmosphere. ADDAM v1.4.2 was used to analyze the projected off-site doses resulting from each of Case 1, Case 2, and Case 3. The ADDAM v1.4.2 qualification report [13] documents all qualification activities which have previously been performed to demonstrate the validity of the code.

ADDAM divides the area surrounding the Bruce Power site into a mesh comprised of concentric circles and 16 equal sectors. Figure 5 illustrates the mesh used in the simulations. ADDAM sectors 1, 2, and 10-16 were excluded from the analysis as they are largely off-shore within 50 km of the Bruce Power site. This is appropriate for the purposes of this study as these sectors are not populated by members of the public. As a result, it would not be necessary for any off-site protective actions to be taken in these areas.

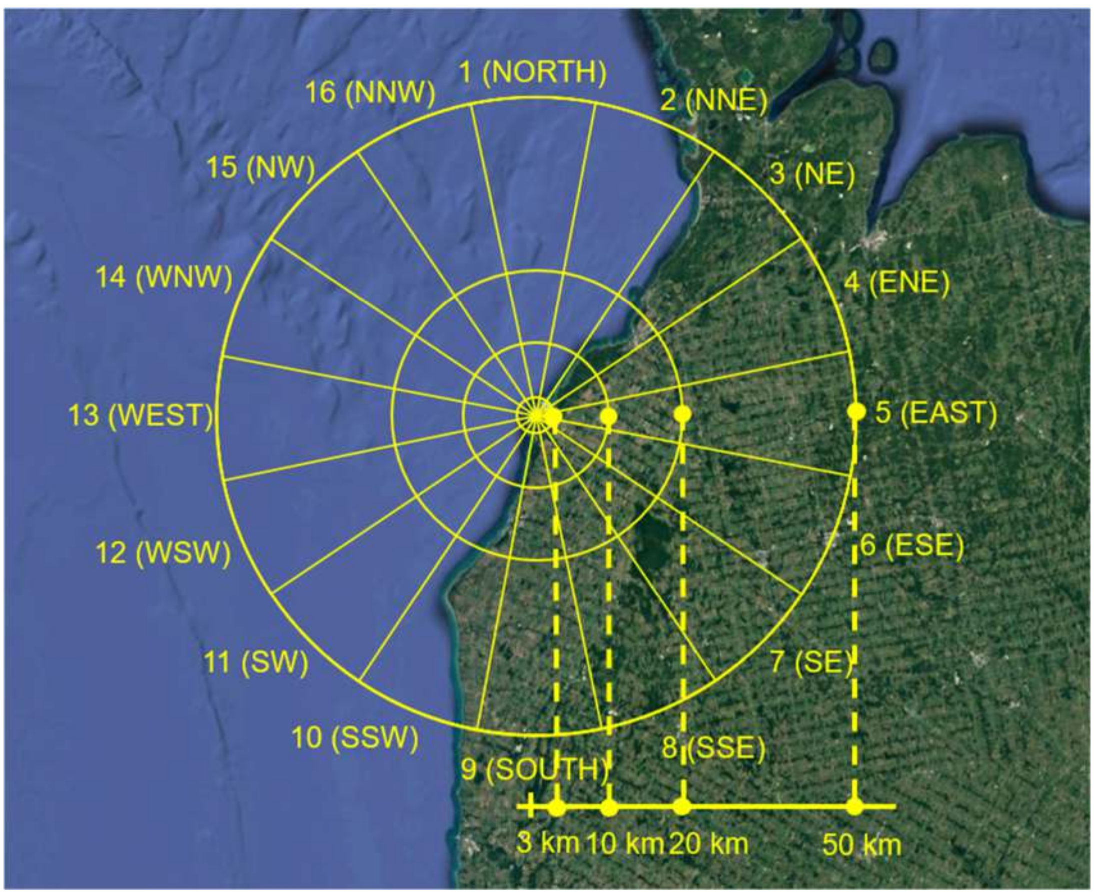

Figure 5: Illustration of Mesh Used for ADDAM Analysis

ADDAM models phenomena related to atmospheric dispersion and dose, which include:

- Plume rise;
- Stack downwash and building entrainment;
- Plume transport and diffusion;
- Wet and dry deposition;
- Plume depletion;
- Radioactive decay and buildup following release;
- External exposure due to cloudshine and groundshine; and
- Internal exposure due to inhalation.

ADDAM simulates these phenomena through the use of a Gaussian plume model, in which material in the plume is assumed to be normally distributed in the vertical and lateral directions. Atmospheric transport is assumed to follow a straight line at a constant wind speed and direction in space and time. This assumption limits the use of Gaussian plume models to downwind distances up to 50 km. At distances beyond 50 km, use of more advanced atmospheric dispersion models such as Gaussian puff or Lagrangian particle dispersion models is required. However, given the focus of this study is on off-site releases out to 50 km, the use of ADDAM is appropriate for this application. Gaussian plume models are used routinely for regulatory applications such as safety analyses, due to their inherent conservatisms. Extensive validation activities, as described in Reference [14], have been performed to demonstrate the ADDAM code overpredicts air concentrations (by a factor of 2 on average) and ground concentrations (e.g., by an order of magnitude or more for dry deposition of iodine) relative to observations.

ADDAM analysis was performed for each of the three accident scenarios considered in the scope of this study. Outputs from MAAP5-CANDU were combined with the following information in order to estimate off-site public doses following the release of radionuclides to the atmosphere:

- Site data (e.g., terrain cover, meteorological roughness length, population data, protection/shield factors);
- Meteorological data (e.g., wind direction, wind speed, precipitation rate, stability class); and
- Radionuclide data (e.g., decay data, dose conversion factors, deposition coefficients).

<u>Site Data</u>

ADDAM is capable of modelling simple variations in terrain cover and roughness between different sectors but is not capable of modelling local variations within an individual sector. The ADDAM model reflects information contained in the Bruce B Safety Report regarding different types of terrain (e.g., forest, grass, soil) on site and surrounding the site [15]. In addition, the ADDAM model reflects the layout of buildings on the Bruce Power site which may have the potential to influence the dispersion of the release [15].

Specific to the Bruce Power site, it is recognized that the shoreline bluffs have the potential to significantly impact off-site dose results. The effect of any obstacle (i.e., bluff or structure) will be dependent on the difference between the length scales of the plume and the obstacle. It is expected that the shoreline bluffs will have little effect on air and ground concentrations as the vertical extent of the plume is expected to be on the same order or greater than the height of the bluffs under neutral stability conditions. Since these conditions approximate potential flow, the ground level concentrations are not expected be significantly affected as the plume travels over the bluffs. Under these conditions, the deformation process is not expected to significantly affect concentrations. However, there is the potential for the bluffs to significantly impact the wind direction and plume location under stable conditions. These potential impacts are addressed through the use of site meteorological data, as described below, and reporting results based on the sector with the highest off-site projected doses.

<u>Meteorological Data</u>

The ADDAM model assumes the plume is dispersed in a straight line at a constant wind speed. In order to address this limitation, site meteorological data is used to run a large number of simulations to analyze a range of potential outcomes which could occur depending on the meteorological conditions during a release. Specifically, ADDAM uses actual historical meteorological data, time-averaged at hourly intervals, recorded over a period of five years at the facility, to generate a cumulative frequency distribution of end points. For this analysis, site meteorological data over a five-year period from 2009 to 2013 was used [15]. This is the most recent five-year period of data that is of sufficient quality to be used for dispersion and dose assessments [16]. The meteorological data from this period is appropriate to use given that prevailing weather conditions surrounding the Bruce Power site have not changed significantly since 2013.

<u>Radionuclide Data</u>

The sources of the radionuclide data were those recommended in Canadian Standards Association (CSA) N288.2:19 [17].

5.2.2.2 Methodology and Assumptions Specific to the Current Study

ADDAM was used to analyze off-site doses in order to address the four questions raised in Section 4.0. This required producing estimates for the following end points of the assessment:

- Effective dose in the first 2 days;
- Effective dose in the first 7 days;
- Thyroid dose in the first 7 days; and
- Ground deposition of I-131 and Cs-137 in the first 7 days.

The public dose reported at a given off-site distance for each of the doses listed above was determined utilizing the following key assumptions:

- The effective dose to an adult receptor is reported as recommended by CSA N288.2:19 for emergency planning evaluations [17].
- The 50th percentile (i.e., median) dose or ground deposition is reported as recommended by CSA N288.2:19 for deterministic calculations to support planning zone sizing [17].
- The highest 50th percentile dose or ground deposition across all 16 compass sectors is reported at the given distance, excluding those sectors that are on water.

6.0 RESULTS

The source terms associated with each accident scenario were input into the ADDAM code in order to estimate off-site doses resulting from the release of the source term to the environment. The subsections below discuss the results of the analysis, including additional details on the overall emergency response.

6.1 Case 1: Large Loss Of Coolant Accident (LOCA)

6.1.1 Summary of Event Response

A large LOCA results in multiple indications in the Main Control Room which immediately alert Operations staff to the abnormal operating condition. This accident is part of the station design basis and there are specific Abnormal Incident Manual (AIM) procedures which are used to direct the Operations response to this event. The AIM procedures have been pre-approved for the response to Design Basis Accidents (DBAs) and are validated prior to being issued to ensure all actions can be readily performed as intended. Additionally, Operations staff are regularly trained on the use of these procedures through the use of a simulator and periodic station drills.

The AIM procedures are structured to ensure measures are in place to maintain the 3 C's of reactor safety:

- Controlling the reactor;
- Cooling the fuel; and
- Containing radiation.

Specific to this event, the 'Control' and 'Cool' functions are addressed by automatic safety systems responding per design. The 'Control' function is addressed by the initiation of one of two independent and diverse shutdown systems. The 'Cool' function is addressed by the automatic initiation of the Emergency Coolant Injection System in response to a decrease in Heat Transport System pressure which results following the LOCA.

Longer-term actions in the LOCA AIM procedure focus on the 'Contain' function. The containment system responds per design and ensures any fission products released from the fuel remain inside the containment structure until a filtered containment vent is required. In this scenario, a containment vent would not be required for several days. When containment conditions are such that a vent is required, a containment vent would occur using the Emergency Filtered Air Discharge System which is an engineered release pathway designed specifically for DBA conditions.

In parallel with performing actions in accordance with the LOCA AIM procedure, Operations staff categorize the event to determine the required level of on-site response and provide appropriate notifications to off-site organizations. The large LOCA scenario analyzed in this study would initially be categorized as an Abnormal Incident which would activate Bruce Power's Emergency Response Organization to aid in co-ordinating the overall event response. In response to receiving event notifications, the Provincial Emergency Operations Centre would be expected to operate in Enhanced Monitoring mode. Based on the initial categorization of the event and estimated off-site doses summarized in Table 6, it is not expected that this event would result in the recommendation of any off-site protective actions.

6.1.2 Source Term

Per Section 5.2, no new analysis was performed to generate the source term for this scenario as the necessary information was already provided in Reference [11]. The source term information for this scenario is provided in Section A.1 of Appendix A.

6.1.3 Off-Site Consequences

The source term information in Section A.1 of Appendix A was input into the Atmospheric Dispersion and Dose Analysis Method (ADDAM) code in order to evaluate the off-site doses to the public resulting from a large Loss of Coolant Accident scenario. The results from the ADDAM analysis are summarized below in Table 6.

Table 6: Off-Site Dose Consequences for Case 1

Scenario	Distance (km)	Plume Exposure (Inhalation, Immersion, Groundshine)			Ingestion Exposure (Contaminated Food)	
		50th Percentile	50th Percentile	50th Percentile	50th Percentile	50th Percentile
		7 Day Adult Thyroid Dose (mSv)	2 Day Effective Dose (mSv)	7 Day Effective Dose (mSv)	Total Cs-137 Ground Concentration (Bq/m^2)	Total I-131 Ground Concentration (Bq/m^2)
Provincial Nuclear Emergency Response Plan Generic Criteria		Iodine Thyroid Blocking 50 mSv in 7 days	Sheltering 10 mSv in 2 days	Evacuation 100 mSv in 7 days	Root Vegetables: 1.94E5 Bq/m^2 Leafy Greens: 4.1E4 Bq/m^2 Grains: 1.67E4 Bq/m^2	Milk: 1.23E6 Bq/m^2
Case 1	3	Note 1	< 0.1	Note 1	Note 1	Note 1
	10		< 0.1			
	20		< 0.1			
	50		< 0.1			
Note 1: Only the sheltering criterion was evaluated due to the small magnitude of the source term. This is appropriate given that projected doses are significantly below the sheltering criterion and other Generic Criteria reflect higher dose values.						

The results in Table 6 indicate that off-site doses for Design Basis Accidents are expected to be significantly lower than the relevant criteria used in the Provincial Nuclear Emergency Response Plan to determine if off-site protective actions are required. More broadly, the results in Table 6 indicate that when various safety systems function per design, the off-site doses to the public are sufficiently low that it is not anticipated any off-site protective actions would be required. Thus, the size of the existing emergency planning zones is more than adequate for managing the off-site response to Design Basis Accidents.

6.2 Case 2: Loss Of Coolant Accident (LOCA)/Loss of Moderator Cooling/Loss of Emergency Coolant Injection

6.2.1 Summary of Event Response

Similar to Case 1, the initiating event is a LOCA which results in multiple indications in the Main Control Room that alert Operations staff to the abnormal operating condition. This event is initially a Design Basis Accident, with the event response directed by a LOCA Abnormal Incident Manual (AIM) procedure. However, Operations staff would be expected to categorize this event as an On-Site Emergency based on entry to the LOCA AIM procedure and failure of Emergency Coolant Injection. In response to this event, the Provincial Emergency Operations Centre would be expected to operate in Partial or Full Activation. However, it is not expected that the initial event categorization would trigger any default off-site protective actions given that this scenario does not include any containment impairments.

The main difference between Case 1 and Case 2 is that in Case 1 the LOCA AIM procedure response is successful whereas in Case 2 the response is unsuccessful. It is expected that efforts during the initial response will focus on attempting to re-establish capabilities required to support execution of actions in the LOCA AIM procedure (e.g., restoration of Emergency Coolant Injection). However, for the purposes of the analysis it is assumed that such actions are unsuccessful. Similarly, it is assumed that any actions which would be attempted in order to establish alternate means of providing fuel cooling are unsuccessful. It is not expected that a LOCA would result in the unavailability of all potential means to re-establish fuel cooling. However, it is necessary to make this assumption in order for the event to progress to a Severe Accident.

In addition to following the LOCA AIM procedure, Operations staff are also monitoring plant conditions in accordance with the Critical Safety Parameter (CSP) Monitoring and Restoration Guide. In comparison to the LOCA AIM procedure, the CSP Monitoring and Restoration Guide utilizes a symptom-based approach to monitor key conditions of interest within the station. The CSP Monitoring and Restoration Guide is used any time an event-based AIM procedure is entered and utilizes a flowchart to monitor the health of CSPs and key Support Parameters (SPs). Collectively, the CSPs and SPs indicate the adequacy of capabilities to control reactor power, maintain fuel cooling, and contain radioactivity. If a CSP or SP deviates outside of acceptable limits, the CSP Monitoring flowchart directs entry into a corresponding Restoration Guideline which contains various actions intended to restore the CSP or SP to within an acceptable range.

In this particular scenario, the LOCA results in a loss of Heat Transport System (HTS) inventory and Emergency Coolant Injection is unavailable to inject water into the HTS. Due to the inability to restore HTS inventory, Operations staff would observe a declining trend in HTS saturation margin[2] to levels that are outside of an acceptable

[2] Saturation margin is a measure of the additional heat the HTS inventory can absorb before it begins to boil. For example, a saturation margin of 0°C would indicate that the HTS inventory remains in liquid form but would be expected to boil if additional heat is added to the HTS.

range. This decreasing saturation margin provides an indication that the heat removal capabilities of the HTS have been adversely affected. The CSP Monitoring flowchart would prompt Operations staff to enter a CSP Restoration Guideline for low HTS saturation margin. However, in this scenario any attempted actions are assumed to be unsuccessful. Similarly, actions in other CSP Restoration Guidelines which Operations staff would attempt to implement (e.g., restoration of HTS storage tank level, pressurizer level, etc.) are assumed to be unsuccessful.

The inability to restore HTS saturation margin will result in boiling off of water in the HTS and the rejection of heat to the moderator. However, this scenario also involves a loss of moderator cooling. As a result, the moderator temperature will begin to increase to the point where it begins to boil-off, resulting in fuel channels being uncovered. Collectively, these two conditions indicate that the entry conditions for a Severe Accident (SA) have been met. Under these conditions, the CSP Restoration Guideline for HTS saturation margin directs Operations staff to transition to use of the Severe Accident Control Room Guide 1 (SACRG-1), which is used for managing the initial Operations response to a SA.

One of the first actions which is implemented in the SACRG response is requesting the deployment of Emergency Mitigating Equipment (EME) if it is not already in the process of being deployed. In this scenario, it is assumed that EME has not yet been requested as the LOCA AIM procedure does not contain any explicit steps to do so. Thus, it is expected that Operations staff would request the deployment of EME ~2 hours into the event.

Severe Accident Management Guidance entry conditions will be met at approximately the same time that the Emergency Management Centre (EMC) is declared operational. In accordance with the LOCA AIM, the Shift Manager would have categorized the event as an On-Site Emergency and initiated notifications of the Emergency Response Organization. In accordance with the Bruce Power Nuclear Emergency Response Plan [9], event categorization is completed within 15 minutes and the on-call Emergency Response Organization is expected to be capable of being operational at the EMC within 90 minutes of receiving the event notification.

Given the extent of activity taking place on-site and at the EMC in a short period of time, it is conservatively assumed that Operations staff request the EMC to deploy EME 3 hours into the event. This assumption is based on the following considerations:

- SA entry conditions are satisfied ~2 hours into the event and there is frequent monitoring of plant parameters using the CSP Monitoring and Restoration Guide which would identify the need to initiate use of SACRG-1.
- The EMC is expected to be operational within 2 hours of the event. There are no significant barriers which would delay staff in reporting to the EMC given that the impacts of the event are limited to one unit.
- Requesting deployment of EME is one of the first steps in SACRG-1. Thus, EME will be requested in a very short period of time following entry to SACRG-1.

After receiving the request from Operations staff, the EMC will direct emergency response personnel to deploy EME. The target is to be able to deploy EME within 2 hours of receiving a request from the station. This capability has been demonstrated through various field validation activities and station drills/exercises such as the Huron Challenge exercise held in 2012 [18]. The impacts of the LOCA event are limited to inside the station. As a result, there would be no obstacles off-site which could potentially delay the deployment of EME. There are pre-approved guidelines for deploying EME on site, which include routing of fire hoses to engineered connection points on the affected unit. These engineered connection points provide the capability to establish a supply of make-up cooling water to the HTS, moderator, or shield tank.

Given the timing of the accident progression, it is expected that several rows of fuel channels would have collapsed by the time EME is deployed, such that efforts to restore HTS inventory would be expected to be ineffective. It is assumed that EME is deployed to establish make-up to the moderator on the accident unit 3 hours after the deployment of EME is requested (i.e., EME is deployed 6 hours into the event). The assumption of 3 hours represents a conservative assumption given that the capability to deploy EME within 2 hours has been previously demonstrated during the Huron Challenge exercise and that this event would only require the deployment of EME on one unit.

Establishing EME make-up to the moderator allows the calandria vessel to be refilled such that all fuel is covered. Ensuring all fuel in the calandria vessel remains covered allows the moderator to remove any decay heat to prevent further progression of the accident.

6.2.2 Source Term

Per Section 5.2, MAAP5-CANDU was used to generate the source term for this scenario. The source term information for this scenario is provided in Section A.2 of Appendix A.

6.2.3 Off-Site Consequences

The source term information in Section A.2 of Appendix A was input into the Atmospheric Dispersion and Dose Analysis Method (ADDAM) code in order to evaluate the off-site consequences associated with a Loss of Coolant Accident/Loss of Emergency Coolant Injection/loss of moderator cooling scenario. The results from the ADDAM analysis are summarized below in Table 7.

Table 7: Off-Site Dose Consequences for Case 2

Scenario	**Distance (km)**	**Plume Exposure (Inhalation, Immersion, Groundshine)**			**Ingestion Exposure (Contaminated Food)**	
		50th Percentile	**50th Percentile**	**50th Percentile**	**50th Percentile**	**50th Percentile**
		7 Day Adult Thyroid Dose (mSv)	**2 Day Effective Dose (mSv)**	**7 Day Effective Dose (mSv)**	**Total Cs-137 Ground Concentration (Bq/m^2)**	**Total I-131 Ground Concentration (Bq/m^2)**
Provincial Nuclear Emergency Response Plan Generic Criteria		Iodine Thyroid Blocking 50 mSv in 7 days	Sheltering 10 mSv in 2 days	Evacuation 100 mSv in 7 days	Root Vegetables: 1.94E5 Bq/m^2 Leafy Greens: 4.1E4 Bq/m^2 Grains: 1.67E4 Bq/m^2	Milk: 1.23E6 Bq/m^2
Case 2	3	0.8	< 0.1	0.5	9.8E+01	1.8E+05
	10	0.2	< 0.1	0.1	2.8E+01	5.1E+04
	20	0.1	< 0.1	< 0.1	1.8E+01	3.0E+04
	50	< 0.1	< 0.1	< 0.1	8.4E+00	1.4E+04

The results in Table 7 indicate that off-site doses for a representative single-unit Severe Accident are expected to be significantly lower than the relevant criteria used in the Provincial Nuclear Emergency Response Plan to determine if off-site actions are required. More broadly, the results in Table 7 illustrate the effectiveness of Emergency Mitigating Equipment in terminating the accident progression to minimize predicted releases to the environment such that it is not anticipated any off-site protective actions would be required. Thus, the size of the existing emergency zones is more than adequate to manage the off-site response to a representative single-unit Severe Accident.

6.3 Case 3: Four-Unit Loss of Heat Sink

6.3.1 On-Site Response

This scenario involves a loss of heat sinks on all 4 units at Bruce B. A loss of heat sinks implies that power and water required to remove decay heat from each reactor are unavailable. This scenario does not make any specific assumptions about the initiating event which causes the loss of heat sink. Events which have the potential to result in this scenario include grid disturbances resulting in a loss of site electrical power or external events (e.g., major tornado, severe ice storm).

A loss of heat sinks event is a Design Basis Accident and there are pre-approved Abnormal Incident Manual (AIM) procedures in place to respond to this event, similar in structure to procedures which would be used to respond to a Loss of Coolant Accident. The nature of the initiating event will determine which AIM procedure is used to direct the initial event response. However, all applicable AIM procedures have the same underlying objectives. Operations staff would be expected to categorize this event as an On-Site Emergency based on the loss of power and subsequent entry into the Emergency Mitigating Equipment Guidelines (EMEGs). In response to this event, the Provincial Emergency Operations Centre would be expected to operate in Partial or Full Activation. However, it is not expected that the initial event categorization would trigger any default off-site protective actions given that this scenario does not include any containment impairments.

This scenario is not expected to result in a Severe Accident (SA) as Emergency Mitigating Equipment (EME) is designed specifically to mitigate this type of event in order to prevent a SA. However, for the purposes of this study it is necessary to assume additional equipment failures occur in order to evaluate the off-site consequences which could potentially result from a multi-unit SA. Assuming this scenario progresses to a SA requires multiple barriers to be ineffective, as summarized below:

- In the event of a loss of electrical power, Operations staff will attempt to operate dedicated back-up electrical generators. Each station has dedicated Standby Generators and Emergency Power Generators[3] which represent two independent sources of back-up electrical power. If one of these sources of power functions per design, the resulting event will be a Design Basis Accident with off-site consequences comparable to Case 1.
- If neither of the back-up sources of electrical power can be established within the first 40 minutes of the event response, Operations staff initiate the use of the EMEGs. The EMEGs are dedicated guidelines used to deploy and implement EME and manage the on-site response in the event of an extended loss of electrical power. The intent of the EMEGs is to prevent the occurrence of fuel damage and ultimately, prevent the event from progressing to a SA.

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[3] Bruce B has Emergency Power Generators whereas Bruce A has dedicated Standby Diesel Generators. Both sets of generators support similar station functions.

- Per Section 6.2.1, it is expected that EME will be deployed within 2 hours of being requested. This capability has successfully been demonstrated in the 2012 Huron Challenge exercise and this capability is regularly demonstrated through periodic testing activities and the deployment of EME when required during station drills/exercises. Additionally, there is considerable time available (i.e., ~12 hours) before Severe Accident Management Guidance entry conditions would be met. Thus, there is high confidence in the capability to deploy EME to prevent the occurrence of a SA.

In the interim, response efforts in the AIM procedures and EMEGs focus on completing actions which do not require AC power in order to extend the amount of time in which decay heat from the reactors can be removed. In addition, as part of the EMEG response, Operations staff will shed non-essential loads in order to preserve battery life and maintain the capability to monitor plant conditions prior to EME being deployed. If required, field staff are equipped with equipment such as flashlights in order to safely navigate their way to designated locations in the plant in order to perform the necessary field actions. Significant delays in accessing locations inside the station are not anticipated unless the initiating event is due to a large steam line failure or the Powerhouse Emergency Ventilation System (PEVS) fails to automatically activate. Specifically, the following actions are credited as part of the EMEG response:

Opening of Boiler Steam Relief Valves

- The analysis credits opening of the Boiler Steam Relief Valves half an hour into the event. Crediting these actions is consistent with the contents of the applicable AIM procedures, which have been subjected to field validation activities prior to the procedures being issued for use.
- Opening of the Boiler Steam Relief Valves reduces the pressure in the steam generators such that make-up from other sources of water can be established later on in the event. This reduction in pressure also extends the period of time in which the steam generators are capable of removing decay heat.

Deaerator Storage Tank Makeup to Steam Generators

- Reducing steam generator pressure allows make-up from the deaerator storage tank to be credited.
- The deaerator storage tank is located at a higher elevation than the steam generators. Establishing this source of make-up relies on gravity to drain the deaerator storage tank inventory into the steam generators and does not require electrical power. Operations staff will perform this action in accordance with the EMEGs and use of deaerator gravity make-up provides 7 hours of inventory to the steam generators [19].

Emergency Coolant Injection (ECI)

Bruce Power reactors have a gas-driven ECI system which does not rely on electrical power to inject water into the Heat Transport System. During a loss of heat sinks event, ECI is injected into the Heat Transport System from the ECI accumulator tanks to provide an additional temporary source of heat removal from the core. ECI injection is credited to be available 15 minutes into the event, consistent with the applicable AIM procedures. Completing these actions within the first 15 minutes ensures station battery power supplies are available to open the necessary valves to establish ECI injection as activities to shed non-essential loads from the batteries will not have been completed by this time. Thus, the ECI accumulator tanks represents another source of water which aids in delaying core degradation and providing additional time to initiate an alternate source of water make-up.

Deployment of EME

The analysis assumes a significant delay in the deployment of EME in order to allow for the event to progress to a SA. Specifically, EME deployment is assumed to take place 2 hours after core collapse, or ~16 hours into the event. This is an extremely conservative assumption given it has been demonstrated that EME can be deployed within 2 hours of being requested.

Establishing EME make-up to the moderator allows the calandria vessel to be refilled such that all fuel is covered. Ensuring all fuel in the calandria vessel remains covered allows the moderator to remove any decay heat to prevent further progression of the accident.

6.3.2 Source Term

Per Section 5.2, MAAP5-CANDU was used to generate the source term for this scenario. The source term information for this scenario is provided in Section A.3 of Appendix A.

6.3.3 Off-Site Consequences

The source term information in Section A.3 of Appendix A was input into the Atmospheric Dispersion and Dose Analysis Method (ADDAM) code in order to evaluate the off-site consequences associated with loss of heat sinks on four units which progresses to a Severe Accident on all units. The results from the ADDAM analysis are summarized below in Table 8.

Table 8: Off-Site Dose Consequences for Case 3

Scenario	**Distance (km)**	**Plume Exposure (Inhalation, Immersion, Groundshine)**			**Ingestion Exposure (Contaminated Food)**	
		50th Percentile	**50th Percentile**	**50th Percentile**	**50th Percentile**	**50th Percentile**
		7 Day Adult Thyroid Dose (mSv)	**2 Day Effective Dose (mSv)**	**7 Day Effective Dose (mSv)**	**Total Cs-137 Ground Concentration (Bq/m²)**	**Total I-131 Ground Concentration (Bq/m²)**
Provincial Nuclear Emergency Response Plan Generic Criteria		Iodine Thyroid Blocking 50 mSv in 7 days	Sheltering 10 mSv in 2 days	Evacuation 100 mSv in 7 days	Root Vegetables: 1.94E5 Bq/m² Leafy Greens: 4.1E4 Bq/m² Grains: 1.67E4 Bq/m²	Milk: 1.23E6 Bq/m²
Case 3	3	0.2	0.2	2.9	9.2E+02	5.6E+04
	10	< 0.1	< 0.1	0.9	2.7E+02	1.5E+04
	20	< 0.1	< 0.1	0.6	1.5E+02	8.6E+03
	50	< 0.1	< 0.1	0.3	7.5E+01	3.9E+03

The results in Table 8 indicate that off-site doses for a representative multi-unit Severe Accident are expected to be significantly lower than the relevant criteria used in the Provincial Nuclear Emergency Response Plan to determine if off-site actions are required. More broadly, the results in Table 8 illustrate the effectiveness of Emergency Mitigating Equipment in terminating the accident progression to minimize the extent of fuel damage such that it is not anticipated any off-site protective actions would be required. It is also worth noting that the analysis results for Case 3 are more severe in comparison to more probable, multi-unit Beyond Design Basis Accidents where Emergency Mitigating Equipment is successfully deployed to prevent a Severe Accident. Thus, the size of the existing emergency zones is more than adequate for managing the off-site response to representative multi-unit Beyond Design Basis Accidents.

7.0 CONCLUSIONS AND RECOMMENDATIONS

7.1 Conclusions

The current Provincial Nuclear Emergency Response Plan (PNERP) planning basis includes considerations for off-site consequences resulting from a range of accident scenarios, including Design Basis Accidents (DBAs) and Beyond Design Basis Accidents (BDBAs) which progress to a Severe Accident. The planning basis for the PNERP is consistent with current industry practices (e.g., 2021 edition of Canadian Standards Association (CSA) N1600).

This study utilizes a 'best estimate' approach in analyzing the response to BDBAs, which is consistent with current industry best practices as outlined in CSA N290.16. The appropriateness of this approach is further supported by the implementation of design features specifically intended for use in BDBA conditions, the existence of procedures to perform necessary actions in support of the BDBA response, and training activities which demonstrate the capability to effectively implement credited actions.

Previous inputs to the PNERP planning basis relied significantly on analysis of off-site consequences which do not fully reflect recent enhancements implemented by Bruce Power as part of its post-Fukushima response or ongoing Drills and Exercises program. Accordingly, it was expected that the use of previous inputs would result in a conservative approach with regards to establishing emergency planning zone sizes surrounding the Bruce Power site. The analysis performed as part of this study has confirmed this to be the case as each of the three analysis scenarios concluded the following:

- There are no areas which are projected to exceed 10 mSv effective dose in the first 2 days outside the currently established Detailed Planning Zone (DPZ). This confirms that the current DPZ of 10 km is adequately sized.
- There are no areas which are projected to exceed 100 mSv effective dose in the first 7 days outside of the currently established Automatic Action Zone (AAZ). This confirms that the current AAZ of 3 km is adequately sized.
- There are no areas which are projected to exceed 50 mSv adult thyroid dose in the first 7 days outside of the currently established DPZ. This confirms that the current DPZ of 10 km is adequately sized.
- There are no areas projected to exceed 1 mSv per year outside of the currently established Ingestion Planning Zone from the ingestion of root vegetables, leafy greens, grains, or milk. This confirms that the current Ingestion Planning Zone of 50 km is adequately sized.

A common result for the DBA and BDBA scenarios analyzed in this study is that the off-site doses to the public are sufficiently low such that it is not anticipated any off-site protective actions would be required. Of particular note is that various BDBA mitigating measures have resulted in the off-site consequences for representative BDBAs (i.e., Case 2 and Case 3) being reduced to levels comparable to, or below, the limits used for DBAs as illustrated below in Figure 6.

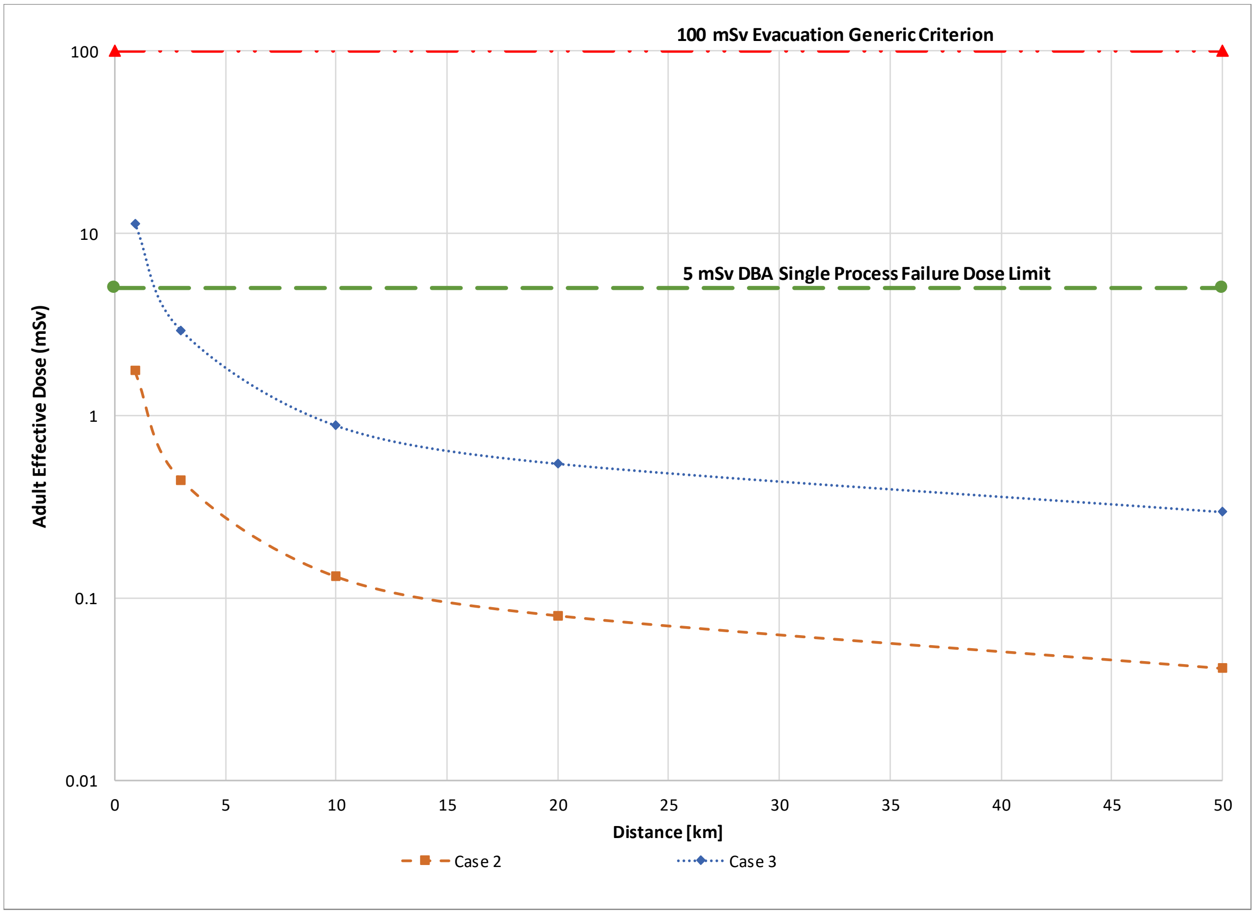

Figure 6: Highest Sector 50th Percentile Effective Dose (mSv) for Adult over 7 Days

The results from the analysis documented in this study are indicative of the following:

- When safety systems designed to mitigate the consequences of Design Basis Accidents perform their credited functions, these systems are highly effective in reducing doses to the public.
- In the unlikely event that permanent plant systems are unavailable to function, Emergency Mitigating Equipment is effective in terminating the accident progression in order to mitigate potential doses to the public. The analysis results for the BDBA cases also illustrate the effectiveness of other post-Fukushima enhancements intended to delay the occurrence of fuel failures and maintain containment integrity.
- The most probable BDBA scenarios are those which do not require off-site protective actions.

The analysis results for Cases 1 to 3 indicate that the existing emergency planning zones are more than adequate to manage the off-site response to the range of accidents which form the PNERP planning basis. The existing emergency planning zones ensure there is margin available to manage the response to higher consequence Severe Accidents which may result in the need for off-site protective actions. Thus, the changes to the emergency planning zones surrounding the Bruce Power site as part of the 2017 PNERP have been confirmed to be, and continue to be, appropriate.

One of the main objectives of Bruce Power's post-Fukushima enhancements was to reduce the likelihood of such events and the implementation of various BDBA mitigating measures has been successful in achieving this objective. With the existing BDBA mitigating measures, Canadian Nuclear Safety Commission safety goals for Large Release Frequency (LRF), which represent the frequency of events which would be expected to result in the need for long-term off-site protective actions, are met for the Bruce A and Bruce B stations. Specifically, recent Probabilistic Safety Assessments for each station demonstrate that the LRF safety goal is met for each station, with predicted LRF values corresponding to lower probabilities than required per the LRF safety goal [20]. Furthermore, Bruce Power will be installing a CFVS at each station in the near future which is specifically designed to reduce the doses to the public resulting from these types of events.

7.2 Recommendations

This analysis was performed to validate the appropriateness of changes to emergency planning zones introduced as part of the 2017 Provincial Nuclear Emergency Response Plan (PNERP) update. Accordingly, this study represents an input which should be utilized in future PNERP updates. Additional details are provided below regarding specific recommendations for the use of this study in future PNERP updates:

1. The analysis undertaken in this study reflects progress Bruce Power has made in implementing post-Fukushima enhancements since the 2017 PNERP update. These enhancements have been analyzed in a manner which is consistent with current industry best practices (i.e., Canadian Standards Association (CSA) N290.16 and CSA N288.2). The results from this study should be used as an input for upcoming revisions to the PNERP Master Plan and PNERP Implementing Plan for BNGS.
2. The study does not credit operation of a Containment Filtered Venting System (CFVS) as this system has yet to be installed at Bruce A and Bruce B. However, Bruce Power has made commitments to the Canadian Nuclear Safety Commission to install a CFVS at Bruce B by the end of 2021 and at Bruce A by the end of 2022. Plans to install a CFVS should be taken into consideration as part of the 2022 PNERP update and further work should be performed in support of subsequent PNERP updates to reflect the capabilities of this system once it has been installed. This is because the CFVS is designed specifically for Beyond Design Basis Accidents and will reduce doses to the public for highly unlikely Severe Accidents which would be expected to result in the need for off-site protective actions.

8.0 REFERENCES

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9.0 GLOSSARY OF TERMS

Automatic Action Zone is a pre-designated area immediately surrounding a reactor facility where pre-planned protective actions would be implemented by default on the basis of reactor facility conditions with the aim of preventing or reducing the occurrence of severe deterministic effects.

Beyond Design Basis Accidents are accident conditions and/or event sequences which are of a relatively low frequency (and hence are not part of the design basis) and which are potentially more severe than Design Basis Accidents. A Design Basis Accident may or may not involve significant core degradation. This includes events with frequencies of occurrence less than 10^{-5} per reactor year.

Contingency Planning Zone is a pre-designated area surrounding a reactor facility, beyond the Detailed Planning Zone, where contingency planning and arrangements are made in advance, so that during a nuclear emergency, protective actions can be extended beyond the Detailed Planning Zone as required to reduce potential for exposure.

Design Basis Accidents are the accident conditions and/or event sequences against which a nuclear power plant is designed and for which the damage to the fuel and the release of radioactive material are known and kept within authorized limits. This includes events with frequencies of occurrence equal to or greater than 10^{-5} per reactor year but less than 10^{-2} per reactor year.

Detailed Planning Zone is a pre-designated area surrounding a reactor facility, incorporating the Automatic Action Zone, where pre-planned protective actions are implemented as needed on the basis of reactor facility conditions, dose modelling, and environmental monitoring, with the aim of preventing or reducing the occurrence of stochastic effects.

Emergency Planning Zones are geographical areas within which predetermined protective action planning is required. The main focus within emergency planning zones is on taking precautionary protective actions, urgent protective actions and other response actions.

Ingestion Planning Zone A pre-designated area surrounding a reactor facility where plans or arrangements are made to:

a) Protect the food chain;

b) Protect drinking water supplies;

c) Restrict consumption and distribution of potentially contaminated produce, wild-grown products, milk from grazing animals, rainwater, animal feed; and

d) Restrict distribution of non-food commodities until further assessments.

Severe Accidents are a subset of Beyond Design Basis Accidents where there is potential for a large release of radioactive materials (i.e., in excess of regulatory limits) due to the following:

- Significant fuel and/or core degradation has occurred,
- Radioactive materials have been released into the containment envelope, and
- Containment failure has or could occur.

Appendix A: SOURCE TERM RESULTS

A.1 CASE 1: LOSS OF COOLANT ACCIDENT SOURCE TERM

The large break Loss of Coolant Accident that results in the greatest extent, duration, and severity of fuel overheating within the core represents the worst case in terms of fission product releases for this category of accident. Note that the fission products generated and discharged during a Design Basis Accident vary over time and are presented in terms of short and long-term releases, and the same representative isotope can be presented multiple times to track the different behaviours of the different forms of the radionuclide.

The Design Basis Accident fission product release source term data described above for a Bruce NPP are presented in Table A-1 and Table A-2.

Table A-1: Bruce NPP Short-Term Fission Product Releases

Representative Isotope	Leakage Releases (TBq)	Impairment Releases (TBq)	Total Releases (TBq)
Xe-138	3.68E+00	1.40E+01	1.77E+01
Kr-88	8.26E-01	3.14E+00	3.95E+00
Xe-133	2.61E+00	5.25E+00	7.88E+00
Xe-131m	6.25E-02	9.57E-02	1.58E-01
I-134	5.56E-01	4.70E-01	1.03E+00
I-135	2.91E-01	2.34E-01	5.25E-01
I-133	1.60E-01	1.10E-01	2.70E-01
Te-132	2.97E-01	1.69E-01	4.66E-01
I-131	1.15E-01	5.41E-02	1.69E-01
Cs-137	6.77E-02	1.72E-02	8.49E-02
Ru-106 (Potentially Volatile 1)	4.16E-01	9.30E-02	5.08E-01
Ru-106 (Potentially Volatile 2)	3.80E-01	8.51E-02	4.66E-01
Ru-106 (Potentially Volatile 3)	6.77E-02	1.51E-02	8.28E-02
Ru-106 (Potentially Volatile 4)	1.78E-01	3.97E-02	2.17E-01
Tritium	1.04E+00	9.91E+00	1.09E+01

Table A-2: Bruce NPP Long-Term Fission Product Releases at Specified Time Interval after Initiating Events (hours)

Isotope	Release Magnitude (TBq) over Time Interval (hours)				
	44.0 - 45.0	45.0 - 69.0	69.0 - 168	168 - 250	250 - 2250
I-129	1.27E-12	2.93E-11	1.37E-10	1.15E-10	4.28E-09
I-131	3.42E-05	7.76E-04	3.18E-03	2.13E-03	8.60E-03
I-132	6.82E-05	1.40E-02	1.53E-01	1.37E-01	2.82E-01
I-133	1.25E-05	1.94E-04	1.64E-04	6.09E-06	4.54E-07
I-134	5.88E-20	4.30E-20	0.00E+00	0.00E+00	0.00E+00
I-135	3.89E-07	3.07E-06	2.77E-07	0.00E+00	0.00E+00
Kr-83m	9.56E-06	2.47E-05	1.54E-08	0.00E+00	0.00E+00
Kr-85	1.03E+00	2.25E+01	6.23E+01	2.99E+01	9.83E+01
Kr-85m	1.51E-02	8.37E-02	1.73E-03	0.00E+00	0.00E+00
Kr-87	6.23E-10	8.31E-10	0.00E+00	0.00E+00	0.00E+00
Kr-88	6.30E-04	2.15E-03	4.62E-06	0.00E+00	0.00E+00
Xe-131m	3.43E+00	7.35E+01	1.85E+02	8.15E+01	2.07E+02
Xe-133	1.62E+02	3.39E+03	7.96E+03	3.19E+03	4.19E+03
Xe-133m	3.70E+00	7.11E+01	1.15E+02	2.35E+01	1.04E+01
Xe-135	2.26E+00	2.47E+01	5.19E+00	2.69E-03	0.00E+00
Xe-135m	3.70E-03	3.05E-02	2.68E-03	0.00E+00	0.00E+00
Tritium	1.43E-01	3.42E+00	1.41E+01	1.17E+01	2.85E+02

A.2 CASE 2: LOSS OF COOLANT ACCIDENT/LOSS OF EMERGENCY COOLANT INJECTION/LOSS OF MODERATOR COOLING SOURCE TERM

The initiating event for this scenario is a Loss of Coolant Accident with a loss of Emergency Coolant Injection and moderator cooling such that the event progresses to a Severe Accident. Emergency Mitigating Equipment make-up to the moderator is credited 6 hours into the event. Establishing Emergency Mitigating Equipment results in the calandria being refilled such that the accident progression is terminated.

With the constant boil-off of the moderator the vacuum reserve is depleted and containment pressure continues to rise reaching atmospheric about 4 hours into the event. At this point fission products begin to be released. This continues for the remainder of the 7-day simulation.

Figure A-1 displays the Case 2 pressure profile and I-131 releases over the 7-day simulation. The source term for this scenario is presented below in Table A-3.

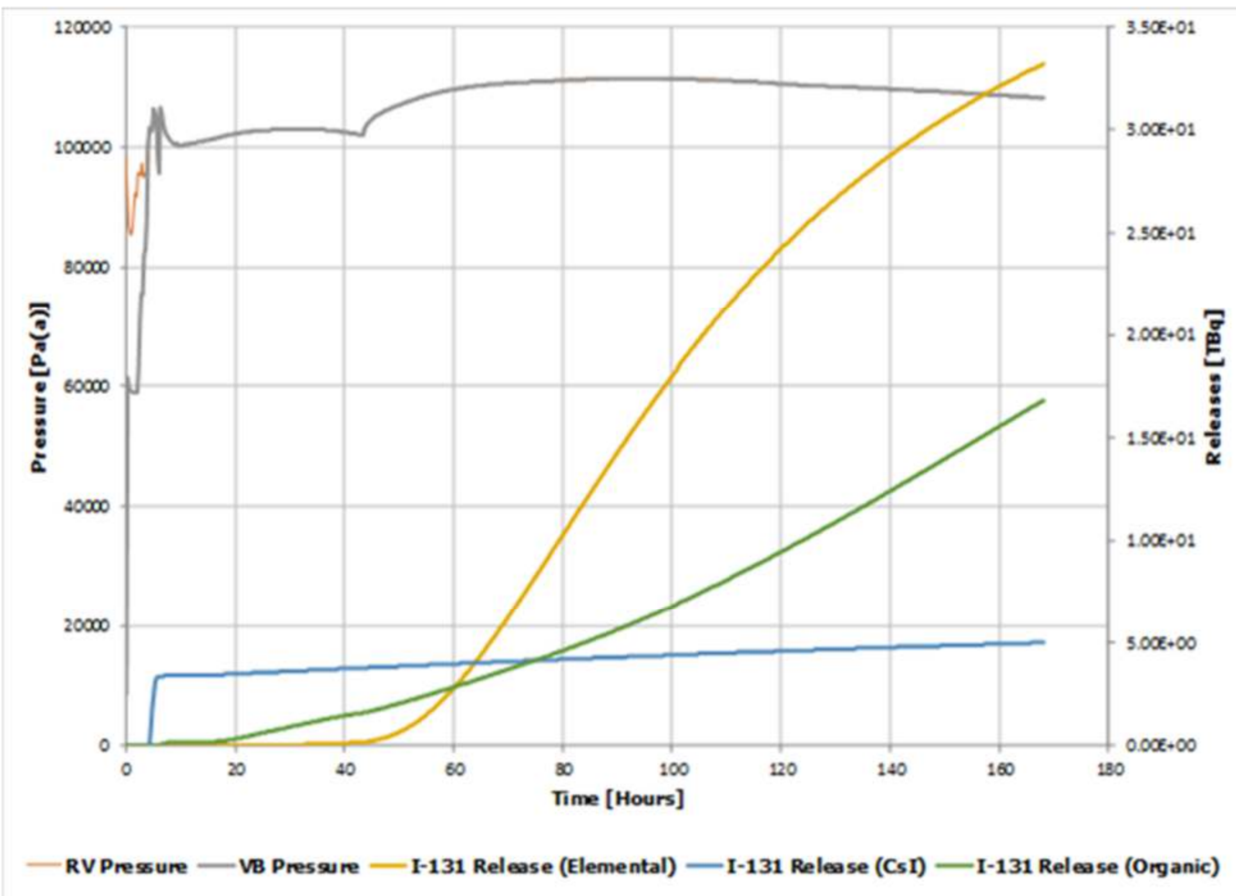

Figure A-1: Case 2 Pressure Profile and I-131 Releases

Table A-3: Source Term for Case 2

Nuclide[4]	TBq[5]		Nuclide[4]	TBq[5]		Nuclide[4]	TBq[5]
HTO	5.97E+04		RB-88	1.01E+00		CE-141	1.05E-04
KR-85	1.16E+03		RB-89	1.10E-05		CE-143	6.37E-05
KR-85M	1.60E+03		Y-90	0.00E+00		CE-144	3.47E-05
KR-87	6.80E+02		Y-91	7.83E-05		PU-238	0.00E+00
KR-88	2.30E+03		Y-92	4.33E-05		PU-239	0.00E+00
XE-131M	5.21E+03		Y-93	4.45E-05		PU-240	0.00E+00
XE-133	7.40E+05		ZR-95	9.79E-05		PU-241	0.00E+00
XE-133M	1.58E+04		ZR-97	5.94E-05		NP-239	1.74E-03
XE-135	3.39E+04		NB-95	8.44E-05		LA-140	1.09E-04
XE-135M	3.46E+03		MO-99	5.37E-01		LA-141	2.92E-05
XE-138	7.52E-02		TE-127	2.14E-01		LA-142	0.00E+00
I-131	3.32E+01		TE-127M	2.00E-02		ND-147	3.68E-05
I-132	2.83E+01		TE-129	1.80E-01		PR-143	9.98E-05
I-133	5.54E+00		TE-129M	1.70E-01		AM-241	0.00E+00
I-134	4.80E-04		TE-131M	6.94E-01		CM-242	0.00E+00
I-135	7.51E-02		TE-132	5.90E+00		CM-244	0.00E+00
I-131NV	5.00E+00		TE-134	1.23E+00		KR-83M	6.95E+02
I-132NV	6.40E+00		SB-127	5.67E-02		KR-89	0.00E+00
I-133NV	6.68E+00		SB-129	6.07E-02		RH-103M	5.24E-01
I-134NV	1.89E+00		SR-89	9.17E-04		RH-106	6.91E-02
I-135NV	4.27E+00		SR-90	1.58E-05		TE-131	7.46E-01
I-131HV	1.68E+01		SR-91	5.27E-04		TE-133	7.48E-01
I-132HV	1.41E+01		SR-92	2.34E-04		TE-133M	7.00E-01
I-133HV	3.54E+00		BA-139	2.06E-03		XE-137	1.73E+03
I-134HV	2.32E-02		BA-140	3.05E-02		CS-138	1.45E+00
I-135HV	3.78E-01		RU-103	5.32E-01		BA-137M	4.06E-04
CS-134	3.90E-02		RU-105	1.20E-01			
CS-136	6.48E-02		RU-106	5.90E-02			
CS-137	1.21E-01		RH-105	3.73E-01			
RB-86	1.61E-03		TC-99M	2.55E-01			

[4] The suffix 'NV' represents releases of CsI, 'HV' represents releases of organic iodine, and no suffix represents releases of element iodine.
[5] For presentation in these tables releases less than 1.00E-05 TBq are rounded to zero.

A.3 CASE 3: FOUR UNIT LOSS OF HEAT SINKS SOURCE TERM

The initiating event for this scenario is a loss of heat sinks which results in a Severe Accident on four units. Emergency Mitigating Equipment make-up to the moderator is credited ~16 hours into the event. Establishing Emergency Mitigating Equipment results in the calandria being refilled such that the accident progression is terminated.

Containment pressure rises quickly due to the boiloff of 4 Heat Transport System and moderator inventories and fission product releases begin after 15 hours once debris is present in the core and continues for the remainder of the 7 day simulation.

Figure A-2 displays the Case 3 pressure profile and I-131 releases over the 7-day simulation. The source term for this scenario is presented below in Table A-4.

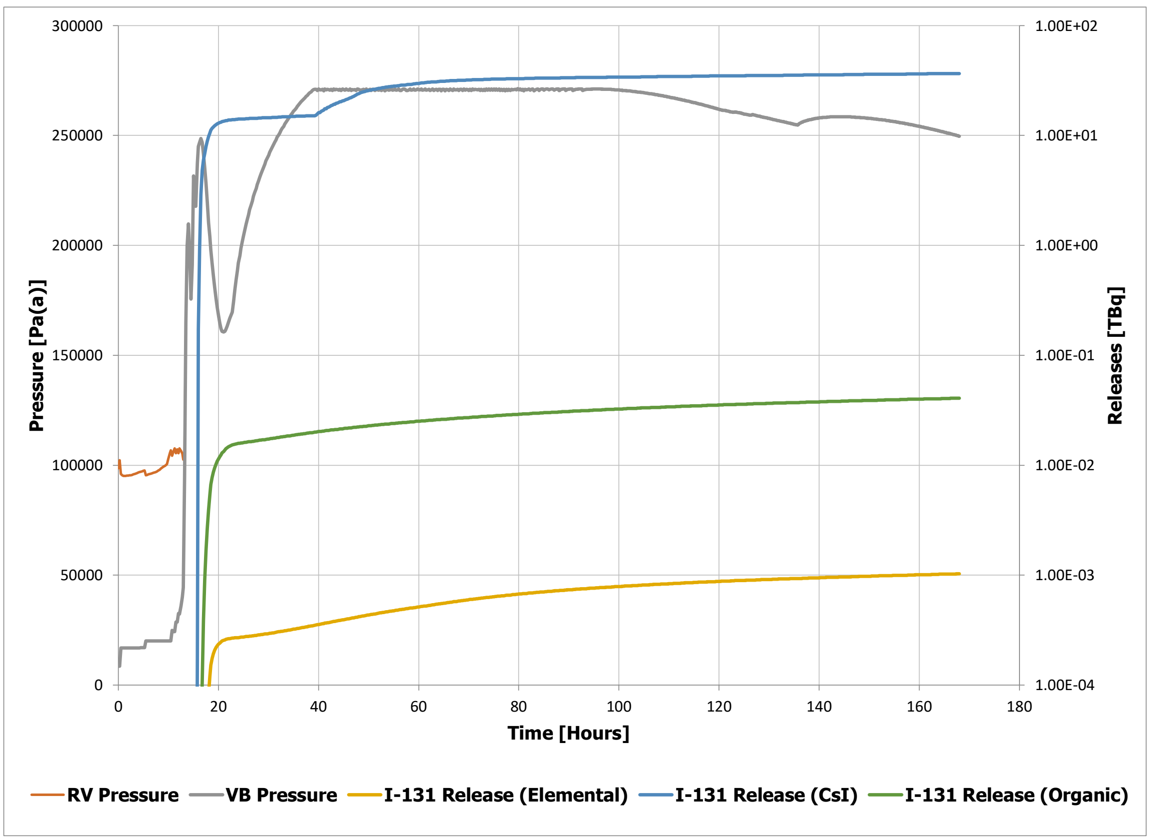

Figure A-2: Case 3 Pressure Profile and I-131 Releases

Table A-4: Source Term for Case 3

Nuclide[6]	TBq[7]		Nuclide[6]	TBq[7]		Nuclide[6]	TBq[7]
HTO	1.61E+0		RB-88	2.01E-01		CE-141	3.49E-02
KR-85	8.58E+0		RB-89	0.00E+0		CE-143	1.71E-02
KR-85M	9.88E+0		Y-90	3.61E-04		CE-144	1.15E-02
KR-87	5.05E-01		Y-91	2.59E-02		PU-238	0.00E+0
KR-88	5.39E+0		Y-92	2.26E-03		PU-239	0.00E+0
XE-131M	3.87E+0		Y-93	6.56E-03		PU-240	0.00E+0
XE-133	5.69E+0		ZR-95	3.24E-02		PU-241	3.02E-04
XE-133M	1.28E+0		ZR-97	1.25E-02		NP-239	5.75E-01
XE-135	4.48E+0		NB-95	2.80E-02		LA-140	3.61E-02
XE-135M	3.16E+0		MO-99	2.24E+0		LA-141	1.04E-03
XE-138	0.00E+0		TE-127	8.14E-01		LA-142	0.00E+0
I-131	1.03E-03		TE-127M	2.47E-01		ND-147	1.20E-02
I-132	1.07E-03		TE-129	9.87E-02		PR-143	3.31E-02
I-133	5.42E-04		TE-129M	2.10E+0		AM-241	0.00E+0
I-134	0.00E+0		TE-131M	4.57E+0		CM-242	0.00E+0
I-135	9.00E-05		TE-132	5.55E+0		CM-244	0.00E+0
I-131NV	2.53E+0		TE-134	3.29E-05		KR-83M	1.83E+0
I-132NV	2.98E+0		SB-127	2.24E+0		KR-89	0.00E+0
I-133NV	2.16E+0		SB-129	1.36E-01		RH-103M	2.85E+0
I-134NV	1.41E-03		SR-89	1.12E-01		RH-106	3.38E+0
I-135NV	4.89E+0		SR-90	1.94E-03		TE-131	1.21E+0
I-131HV	4.08E-02		SR-91	2.77E-02		TE-133	7.49E-05
I-132HV	4.44E-02		SR-92	1.14E-03		TE-133M	3.52E-04
I-133HV	2.66E-02		BA-139	3.17E-04		XE-137	0.00E+0
I-134HV	0.00E+0		BA-140	2.52E+0		CS-138	0.00E+0
I-135HV	4.96E-03		RU-103	2.94E+0		BA-137M	3.41E-02
CS-134	2.03E-01		RU-105	3.44E-01			
CS-136	3.30E-01		RU-106	3.25E+0			
CS-137	6.32E-01		RH-105	1.52E+0			
RB-86	8.28E-03		TC-99M	2.37E+0			

[6] The suffix 'NV' represents releases of CsI, 'HV' represents releases of organic iodine, and no suffix represents releases of element iodine.
[7] For presentation in these tables releases less than 1.00E-05 TBq are rounded to zero.